LO-0616-49507

Enclosure 1:

"The NuScale Design", PM-0616-49510-NP, Revision 0, nonproprietary version

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NuScale Nonproprietary

The NuScale Design

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Revision: 0

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Acknowledgement & Disclaimer

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Agenda (Part 1)

- Design overview
- NuScale Power module
- Helical coil steam generator
- Core and fuel design
- Systems
- Refueling
- Operations
- Electric power systems
- Instrumentation and actuation signals
- Power module piping and valves
- Reactor building crane

Agenda (Part 2)

- Probabilistic risk assessment
- Spent fuel pool and storage rack design
- Ultimate heat sink reactor building pool
- Comparison to a large passive PWR
- Accident response
- DCA test program
- Security by design
- Fire protection
- Radiological protection
- The NuScale DCA

NuScale Power - A 21st Century Company

- **2000** Basic design concept developed
- **2003** Integral test facility first operational
- **2007** NuScale Power formed
- **2008** Began NRC DCA preapplication project
- **2011** Fluor became major investor and strategic partner
- **2012** Twelve-reactor control room simulator commissioned; ARES and Rock Creek Technologies joined as partners
- **2013** Western Initiative for Nuclear formalized
- **2013** Rolls Royce joined as partner
- **2013** NuScale selected to receive major grant from U.S. DOE
- **2014** Enercon joined as partner
- **2015** Ultra Electronics joined as a partner; AREVA becomes fuel designer and supplier
- **2016** As of May 2016:

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NuScale Engineering Offices in Corvallis, Oregon

One-third Scale Integral Test Facility

NuScale Control Room Simulator

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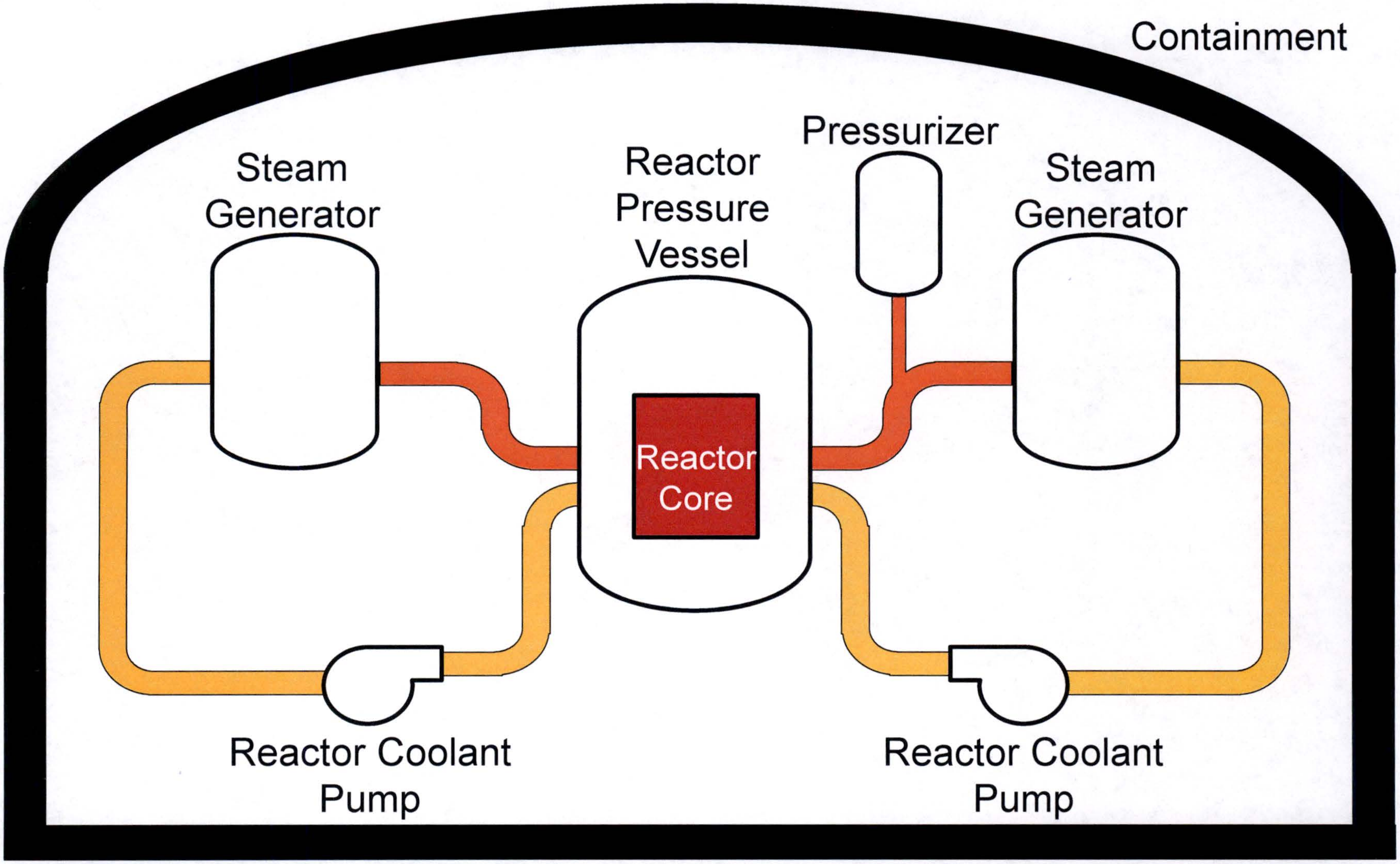
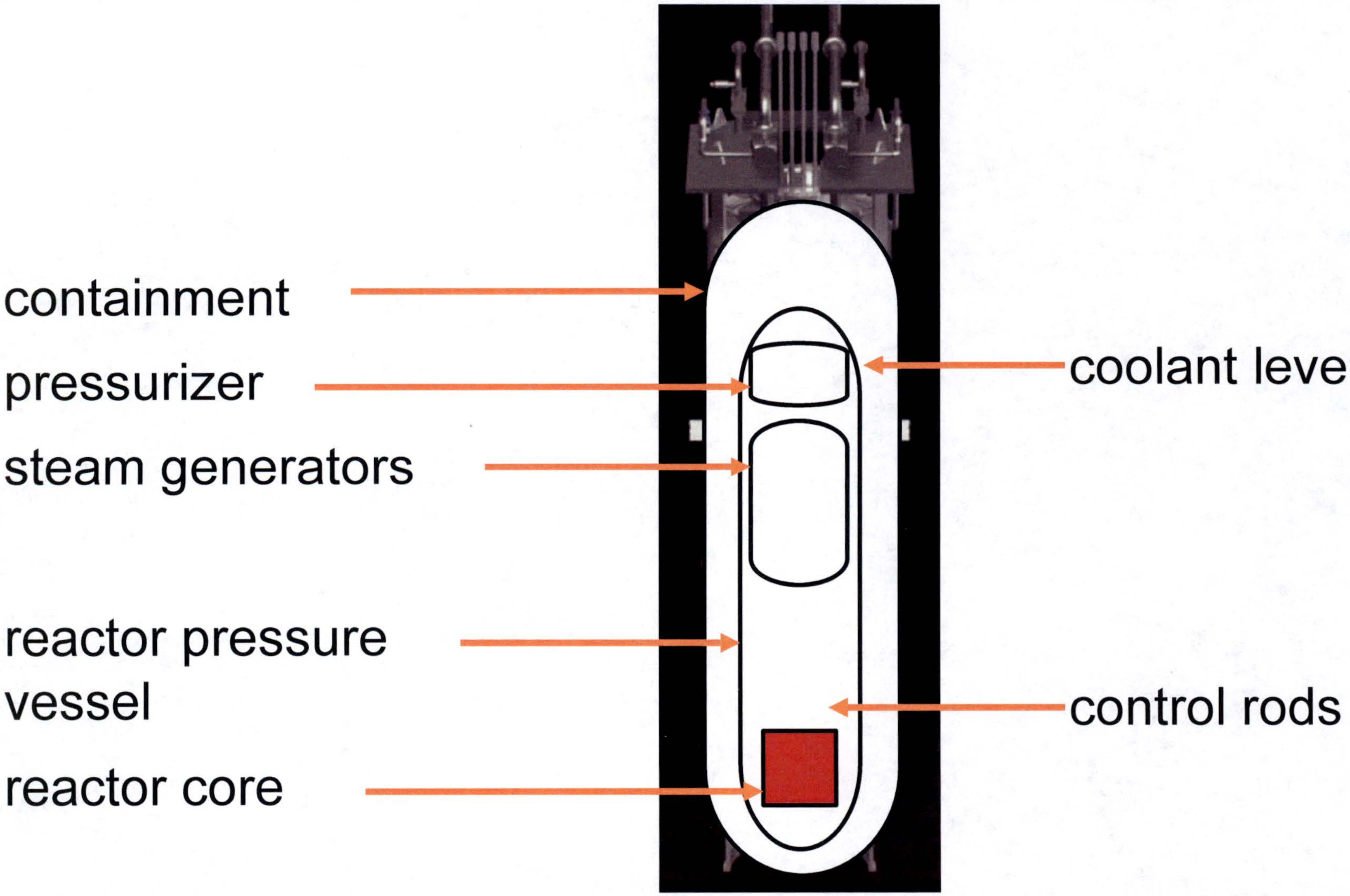
Design Overview

Operating PWR NSSS- Outside the RPV



NuScale PWR NSSS – It's All Inside



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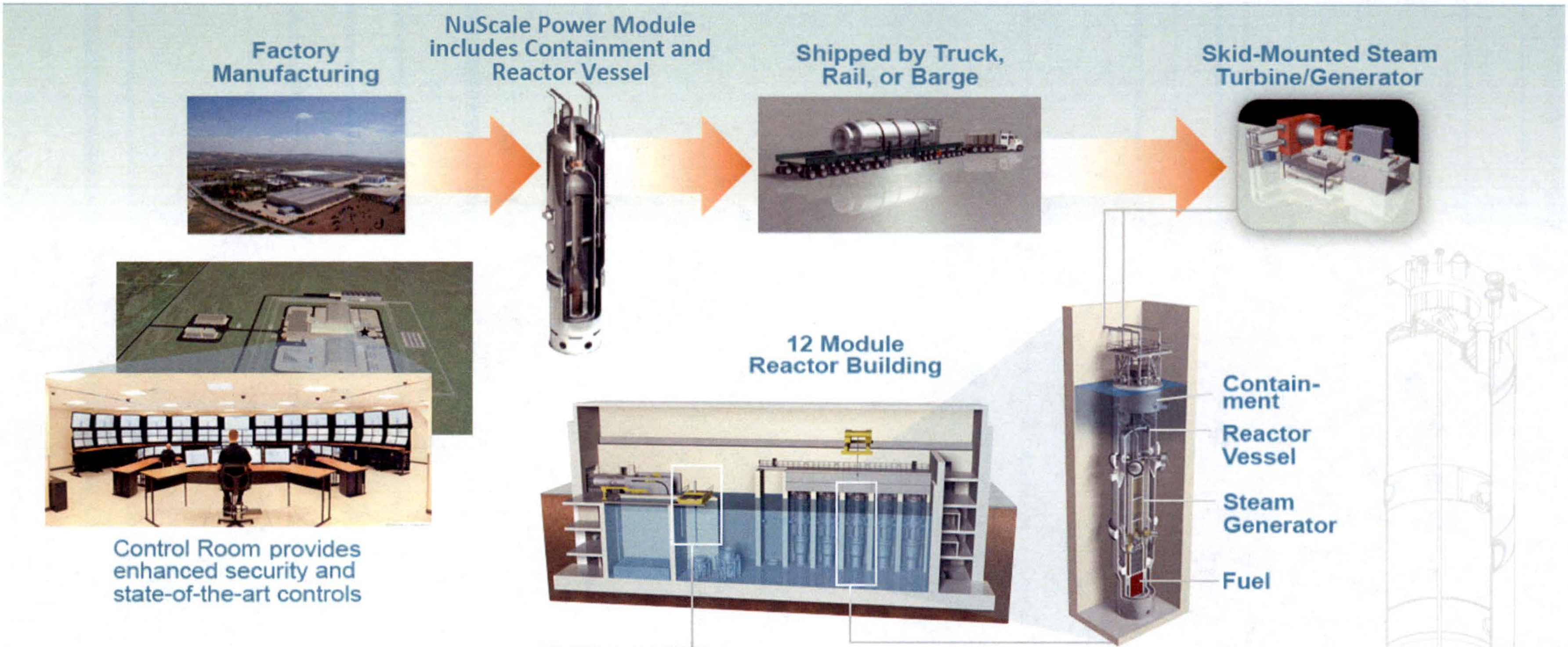
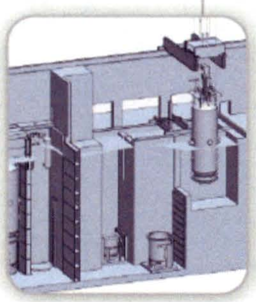
NuScale Scalable Modular Design



Each Module is refueled underwater while the remainder of the plant produces power

- Refueled once every 24 months
- Capable of 48-month fuel cycle
- 10 Day Refueling Target



Each Module Installed in its own Isolated Bay

- Natural Circulation (No Reactor Coolant Pumps)
- 37 Standard 17x17 PWR Fuel (Half-Height) Fuel Assemblies
- Standard Magnetic Jack Control Rod Drives
- Internal Helical Coil Steam Generators and Pressurizer
- 50 MWe Gross Power

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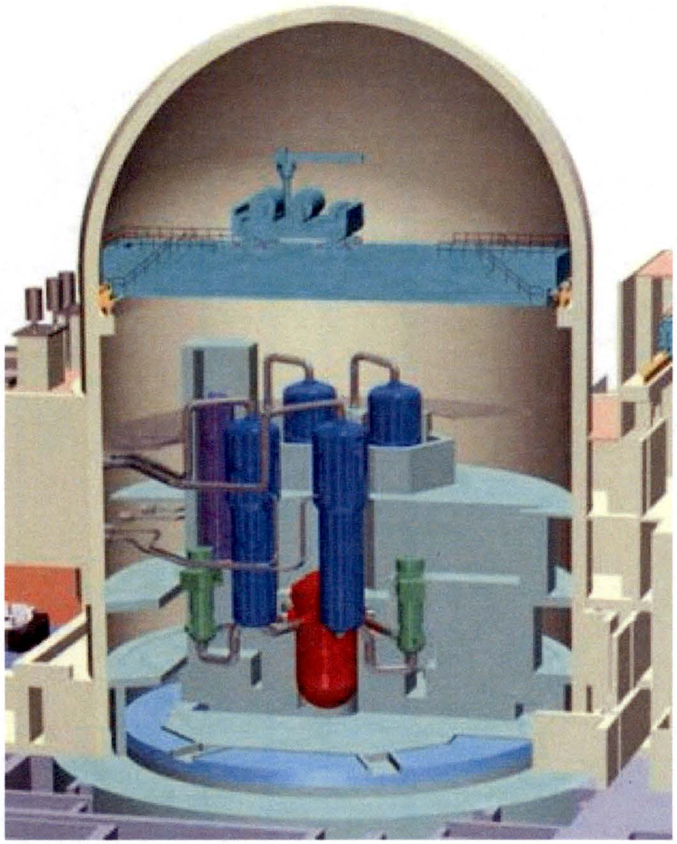
NuScale Design Safety Case

- Fewer, simpler, and passive safety systems
- No core damage for all design basis accidents (DBAs)
- No operator action required for any DBAs
- No AC or DC power required for any DBAs
- Safe shutdown earthquake (SSE) of 0.5 g that is significantly larger than a recently approved passive design

Size Comparison

Typical pressurized-water reactor (PWR) containment and reactor system



*Source: NRC

NuScale Power Module
Combined containment vessel and integral reactor system – about the size of a PWR steam generator

Plant Overview

Overall Plant	
• Net electrical output	Up to 570 MWe (nominal) (600 Mwe gross -30 Mwe house load)
• Plant thermal efficiency	> 31%
• Number of power generation units	Up to 12
• Nominal plant capacity factor	> 95%
• Total plant protected area • Total owner controlled area	~35 acres ~70 acres
Power Generation Unit	
• Number of reactors	One
• Gross electrical output	50 MWe
• Steam generator number	Two independent tube bundles (50% capacity each)
• Steam generator type	Vertical helical coil tube (secondary coolant boils inside tube)
• Steam cycle	Superheated, 500 psia steam
• Turbine throttle conditions	3.3 MPa (475 psia)
• Steam flow	67.5 kg/s (536,200 lb/hr)
• Feedwater temperature	149° C (300 °F)
Reactor Core	
• Thermal power rating	160 MWth (gross)
• Operating pressure	12.7 MPa (1850 psia)
• Fuel design	UO_2 (< 4.95% U^{235} enrichment); 37 half height 17x17 geometry lattice fuel assemblies; AREVA M5 cladding; negative reactivity coefficients
• Refueling interval	24 months (capable of 48 months)

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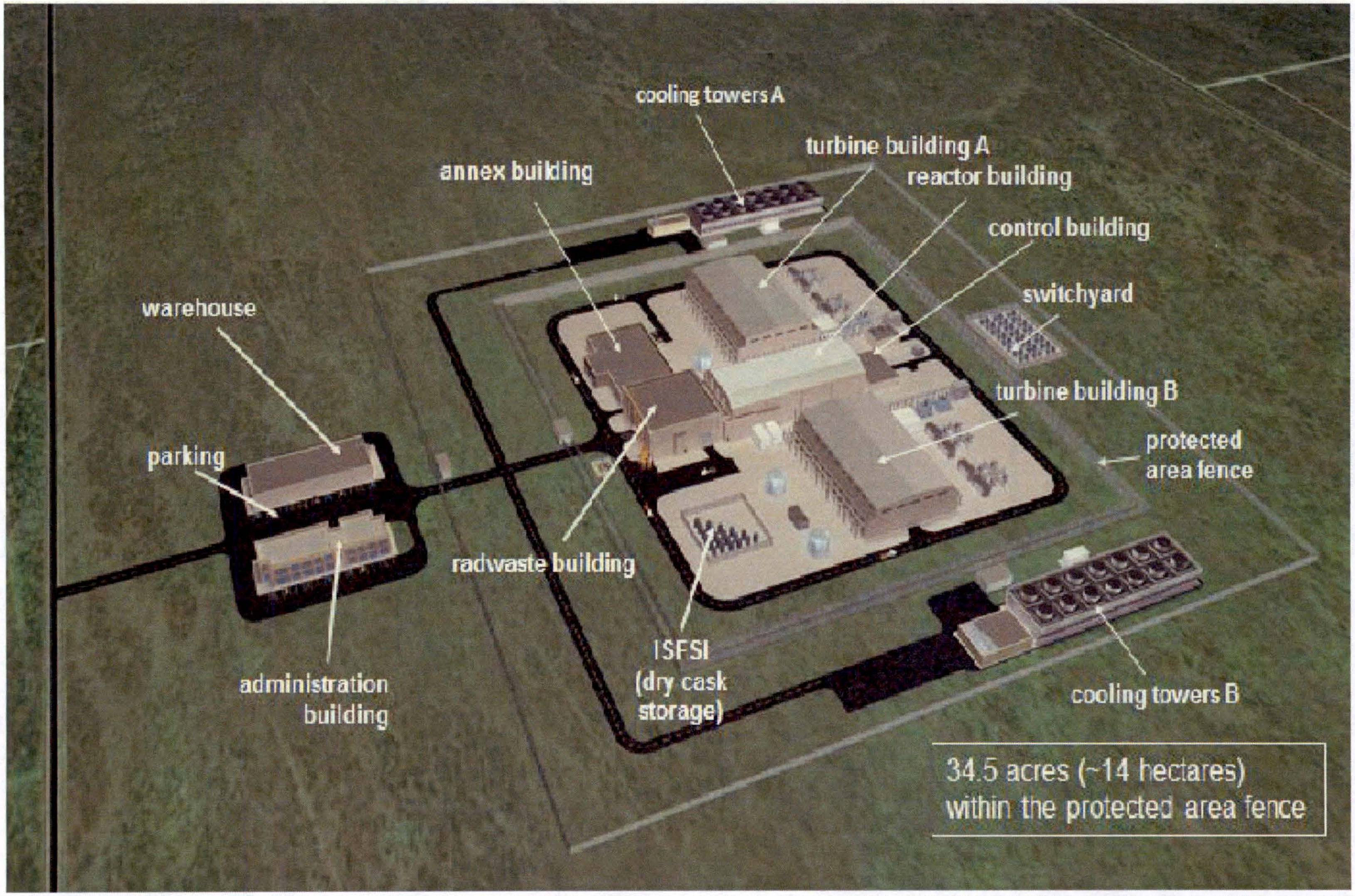
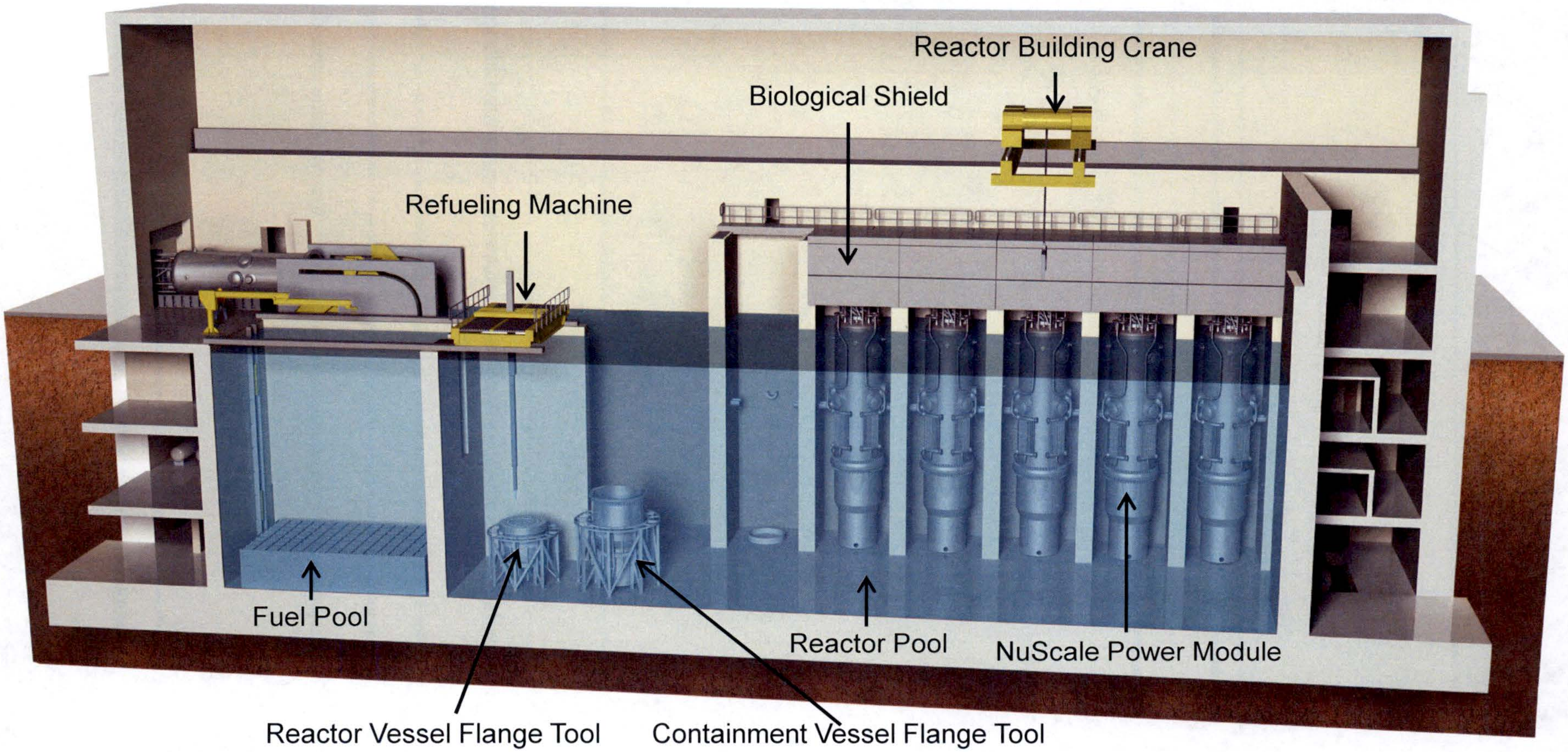
Site Layout


cooling towers A
turbine building A
annex building
reactor building
control building
switchyard
warehouse
turbine building B
protected area fence
parking
radwaste building
ISFSI (dry cask storage)
administration building
cooling towers B
34.5 acres (~14 hectares) within the protected area fence

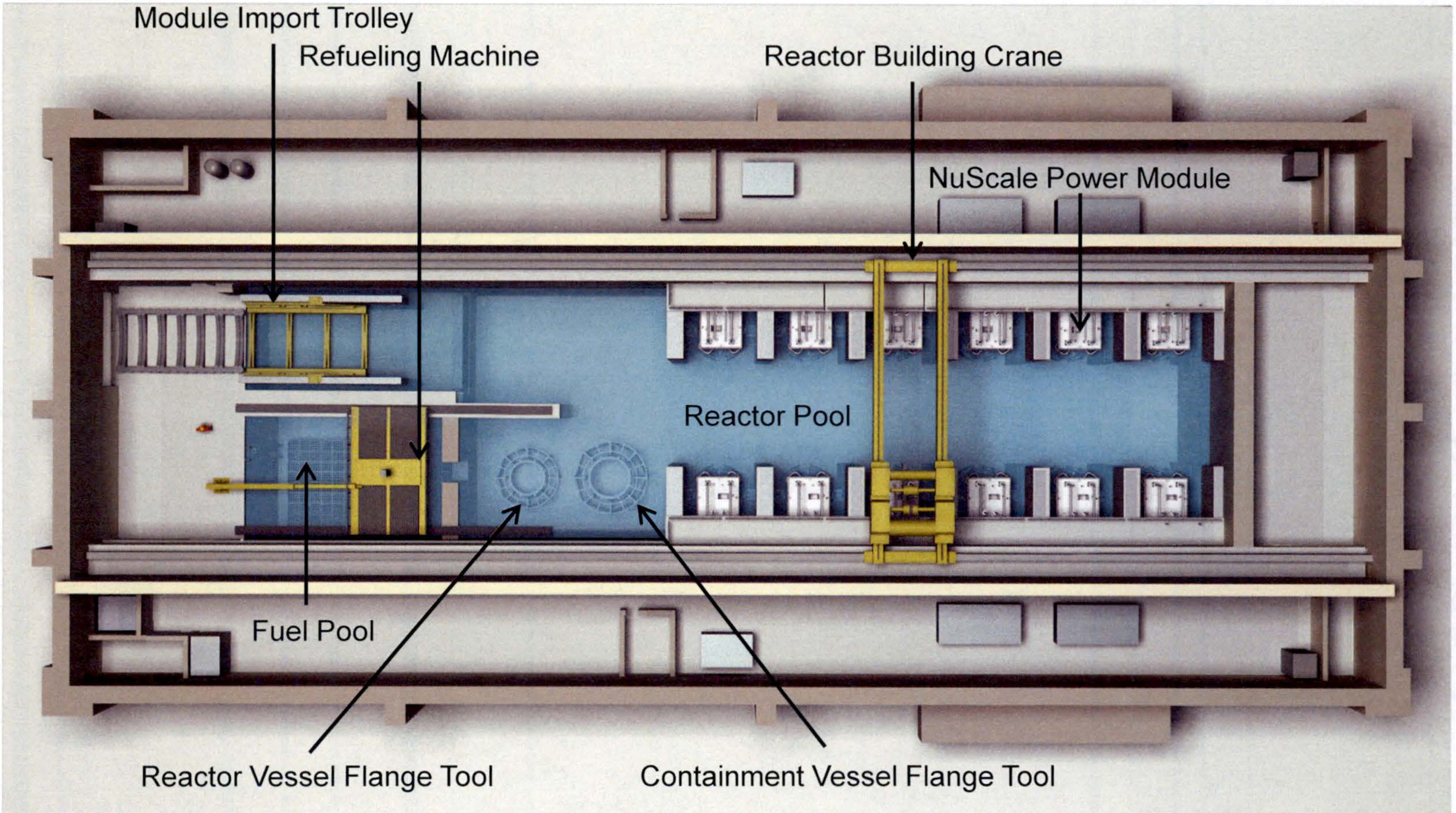
Reactor Building Cross Section

Reactor building houses NuScale power modules, fuel pool, and reactor pool



NuScale Power

Reactor Building Overhead View



Reactor Building

- **Seismic Category I and aircraft impact resistant building**
- **Contiguous reactor module pool, refueling pool, and spent fuel pool (separated from other pools by a weir wall)**
- Major systems:
  - **NuScale Power modules**
  - chemical and volume control system
  - boron addition system
  - non-safety electrical systems
  - **safety I&C systems**
  - nonsafety I&C systems
  - remote shutdown station
  - module import, assembly, and handling equipment

Note: safety related SSCs are highlighted in **bold**

Reactor Building Dimensions

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}}2(a),(c)

- Reactor building contiguous power module, refueling, and new/spent fuel stainless steel lined pool

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Control Building

- **Seismic Category I Building**
  - main control room (MCR) underground
  - central alarm station (CAS)
  - technical support center (TSC) above ground
  - tunnel to reactor building underground connected by airlock adjacent to MCR
  - control room habitability system (72-hour habitability supply of pressurized emergency air bottles) at floor below the MCR

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Radioactive Waste Building

- **Seismic Category II Building**
- Major equipment
  - radioactive waste management systems – liquid, gaseous, and solid radioactive waste
  - module import trolley
  - reactor building and radioactive waste building ventilation

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Turbine Building

- **Seismic Design Per ASCE 7**
- Each building contains 6 turbine generator sets
- Major equipment for each power module
  - turbine
  - generator
  - condensers (100% steam bypass capability)
  - feedwater pumps
  - condensate pumps
  - feedwater heaters

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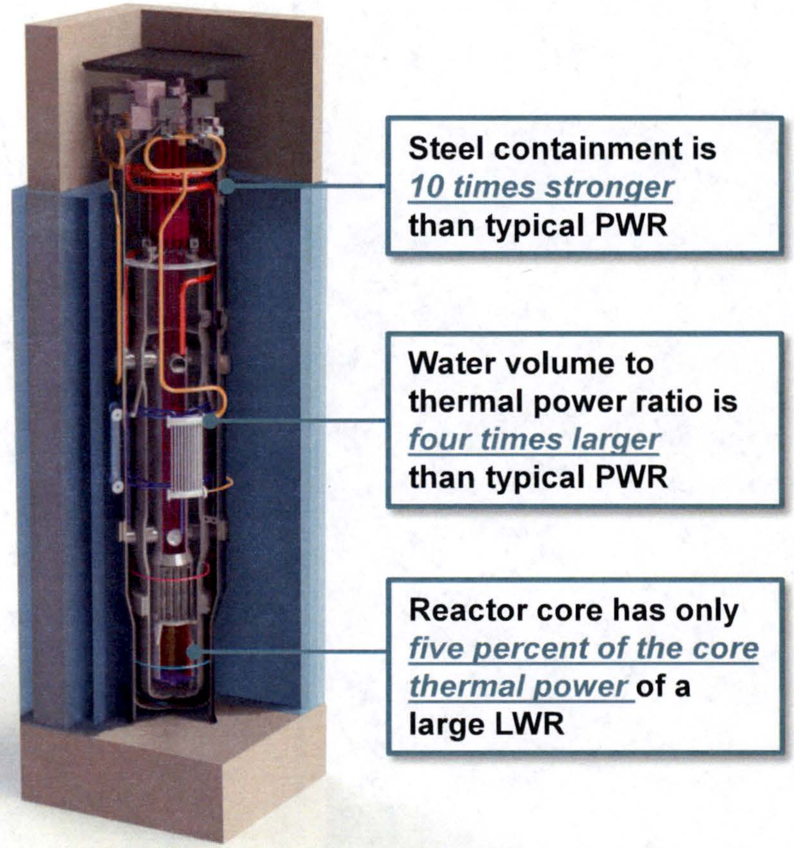
The NuScale Power Module

NuScale Power Module

All safety equipment needed to protect the core is shown on this picture

- A NuScale Power module (NPM) includes the nuclear reactor, steam generators, pressurizer and containment in an integral package that eliminates reactor coolant pumps and large bore pipes (no large break LOCA).
- The NPM is passively safe relying upon natural physics of convection, conduction, and gravity to cool the reactor during normal operation, shutdown, and emergency core cooling (no reactor coolant pumps).
- Each NPM is installed below-grade in a seismically robust building within a steel-lined reactor pool.
- Each NPM is 50 MWe gross and factory built.
- NPMs can be incrementally added to match load growth (up to 12 NPMs for 570 MWe total net output).



160 MWt Reactor Power Module

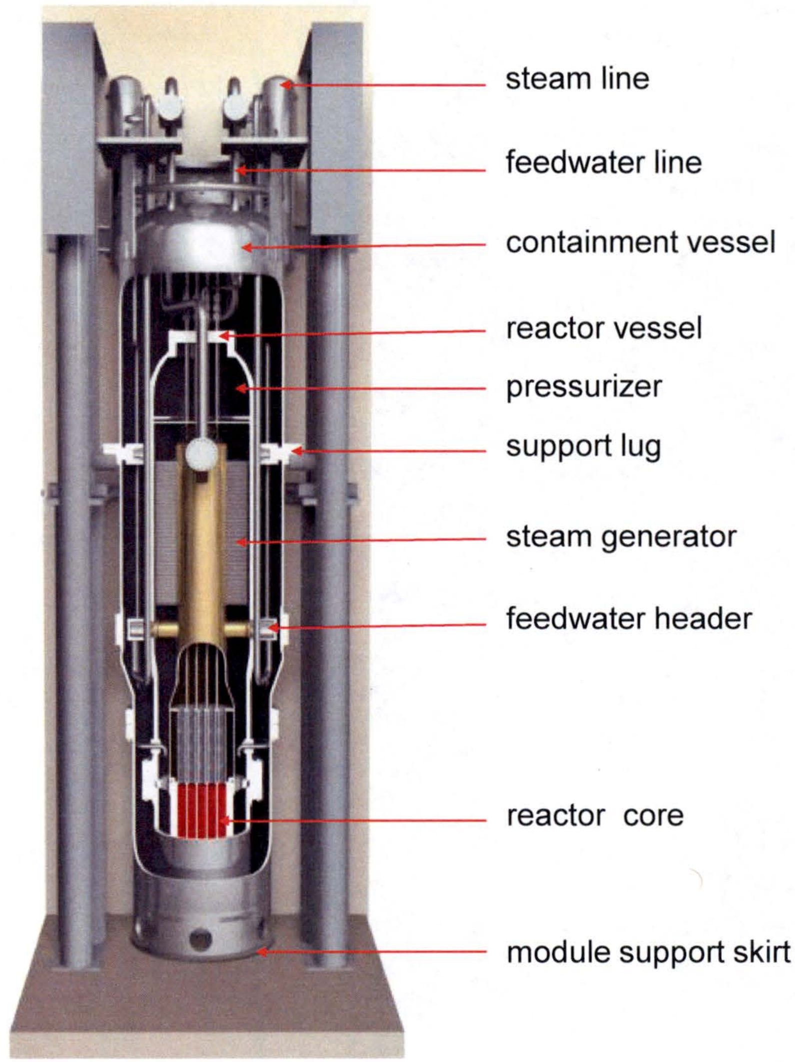
Power Module Overview

Natural convection for cooling

- Passively safe, driven by gravity, natural circulation of water over the fuel
- No reactor coolant pumps, no need for emergency generators

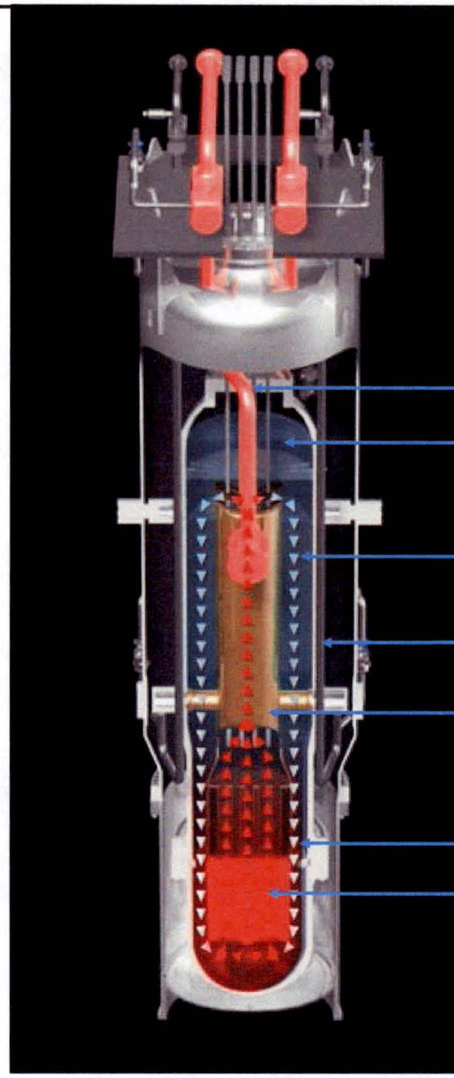
Simple and small

- Reactor is 1/20th the core thermal power of large reactors
- Integrated reactor design, no large-break loss-of-coolant accidents



Natural Circulation Operation

- Integrated reactor vessel
  - steam generator, pressurizer, fuel inside a single vessel
- Natural circulation flow
  - No reactor coolant pumps
  - No external power
- Helical coil steam generator
  - Designed to maximize thermal efficiency under low flow conditions



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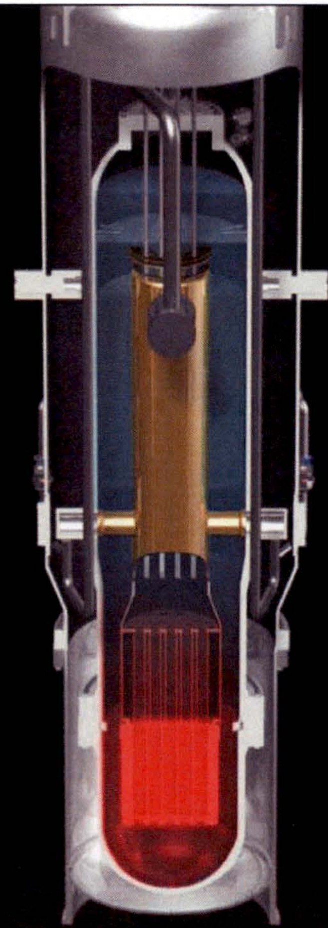
NuScale's Normal Operation

Flows to the downcomer

Flows up the riser

Heated in the core



Transfers heat to steam generator

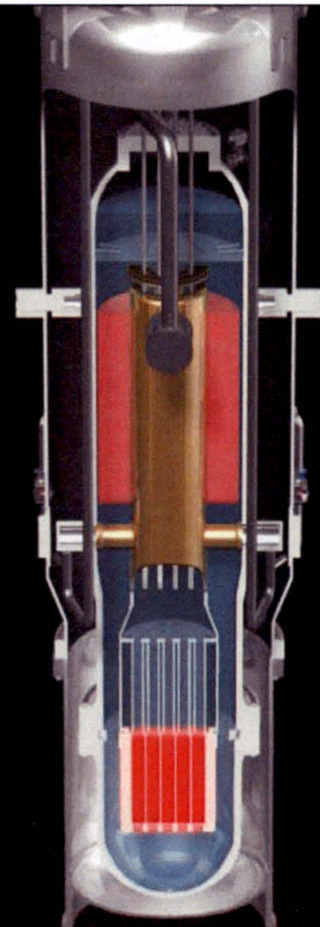
Flows to bottom of reactor pressure vessel

Repeats the cycle

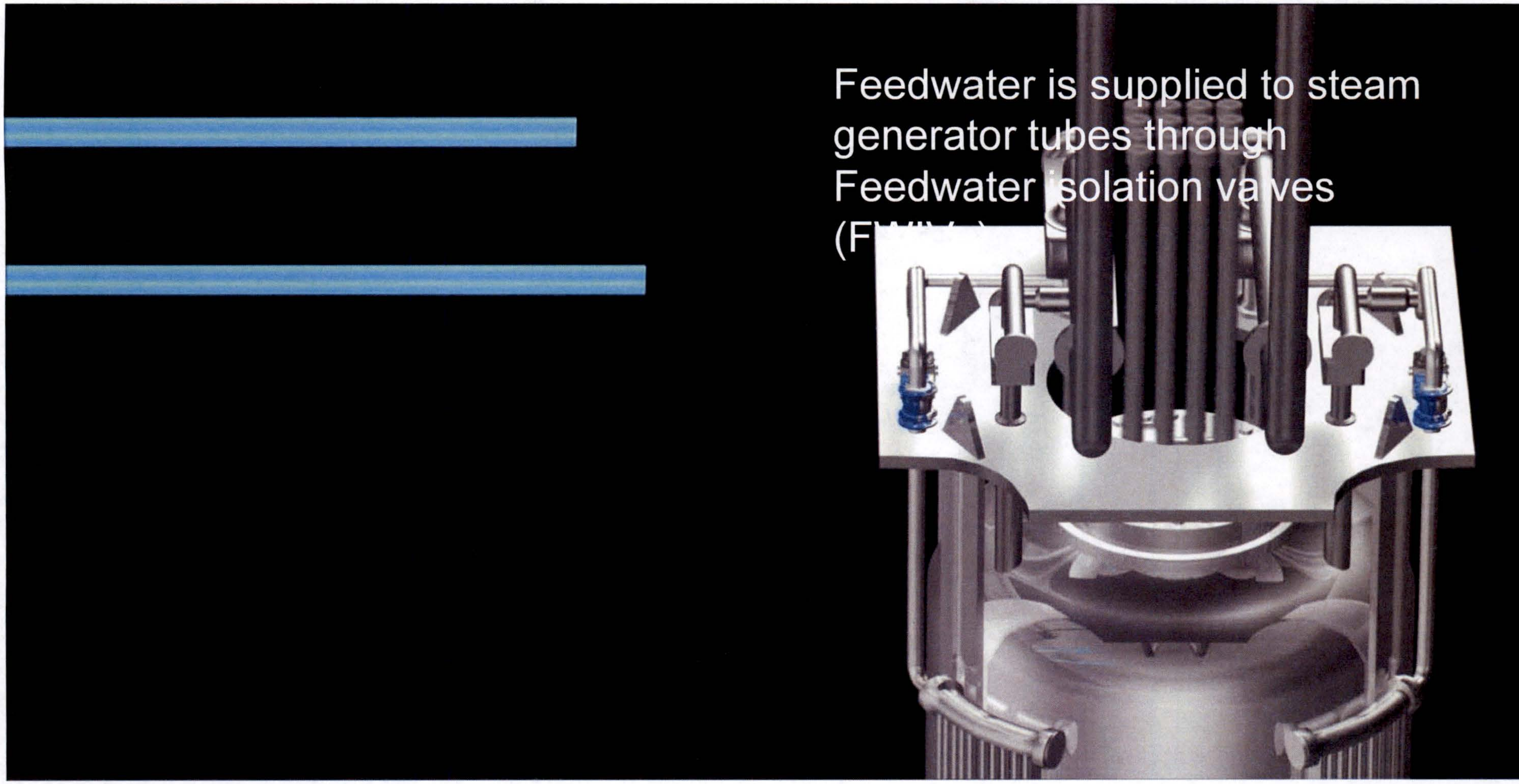
NuScale's Normal Operation

Steam flows out through main steam isolation valves (MSIVs)

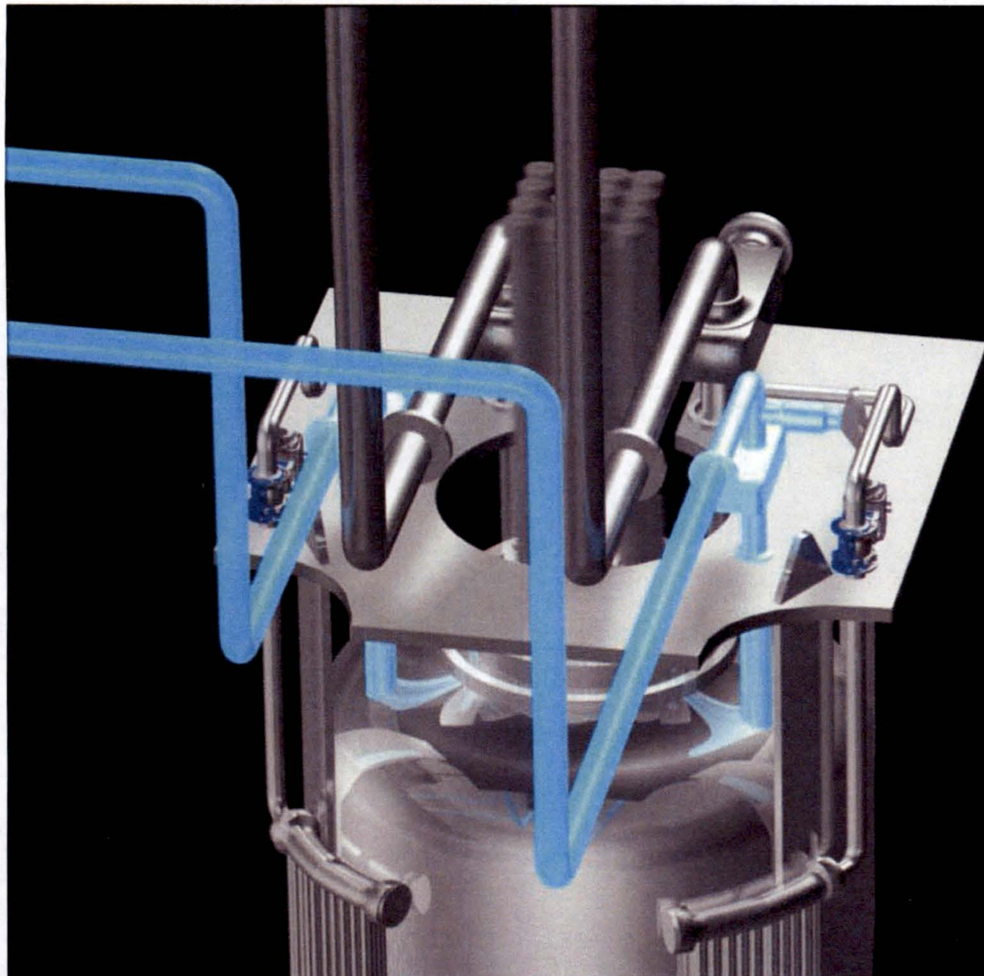
Feedwater becomes steam inside steam generator tubes



NuScale's Normal Operation



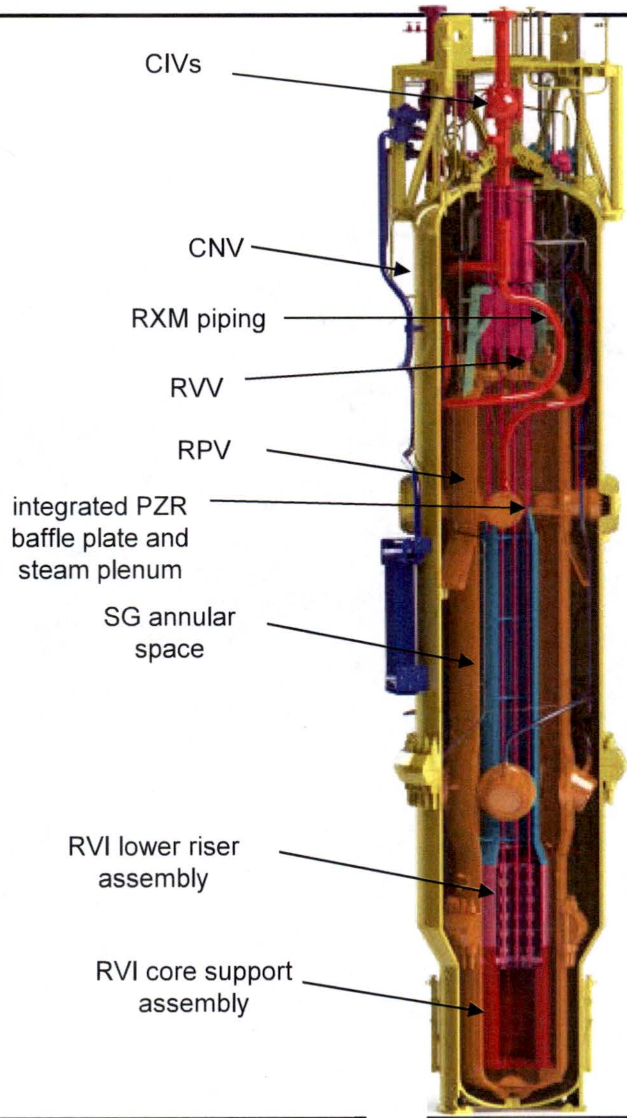
NuScale's Normal Operation



Feedwater is again heated through steam generator tubes by reactor coolant

Repeats the cycle

NuScale Power Module



CIV = containment isolation valve

CNV = containment vessel

PZR = pressurizer

RPV = reactor pressure vessel

RRV = reactor recirculation valve

RVI = reactor vessel internals

RVV = reactor vent valve

RXM = reactor module

SG = steam generator

Parameter	Numerical Value
CNV height; OD max./min.	75.8 ft and 15/11.2 ft
RPV height and OD	58 ft. and 10 ft.
Module weight (metal)	762 tons
RPV weight (metal)	343 tons (w/o fuel)

Power Module Arrangement

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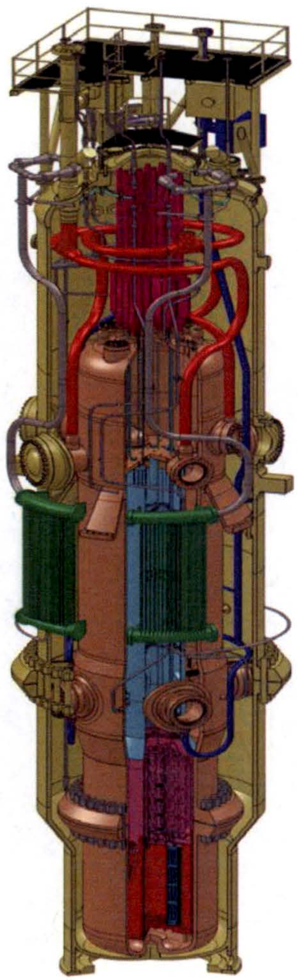

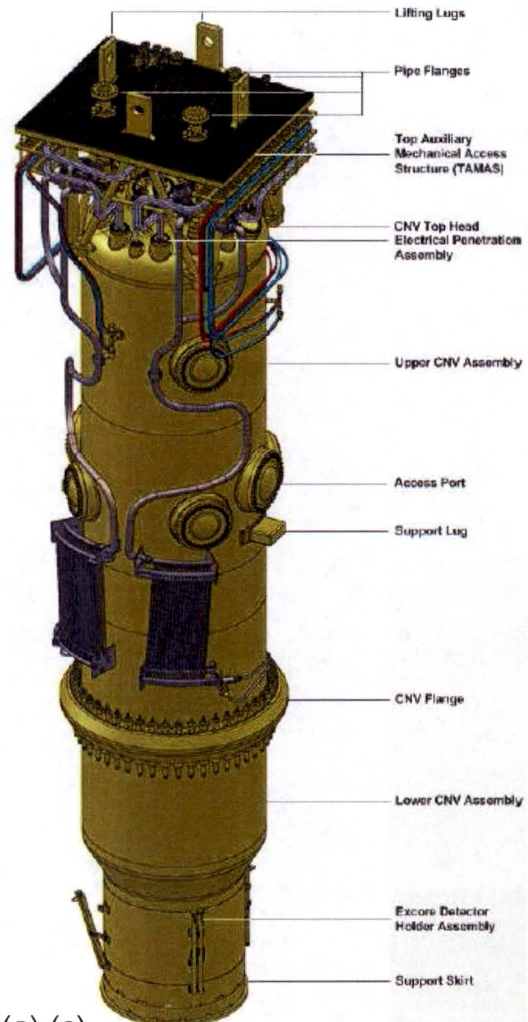

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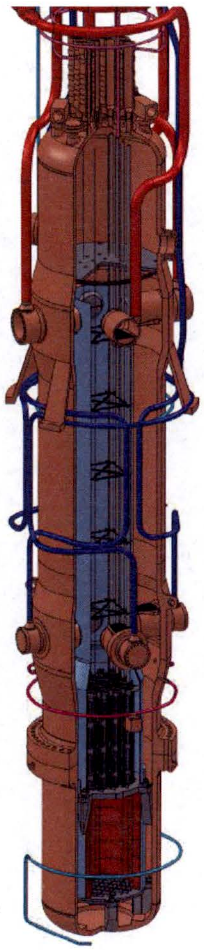
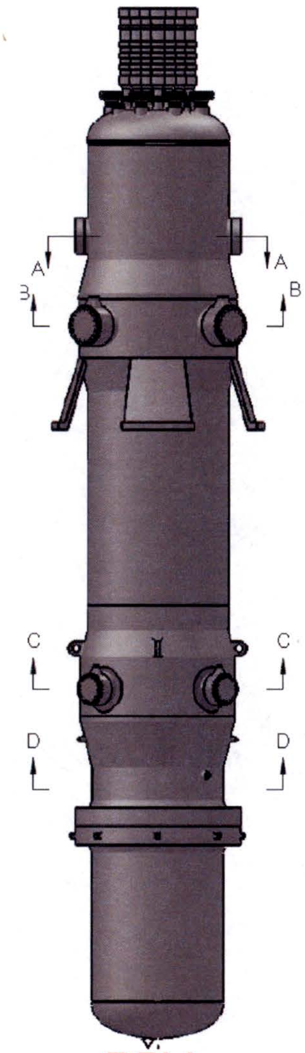
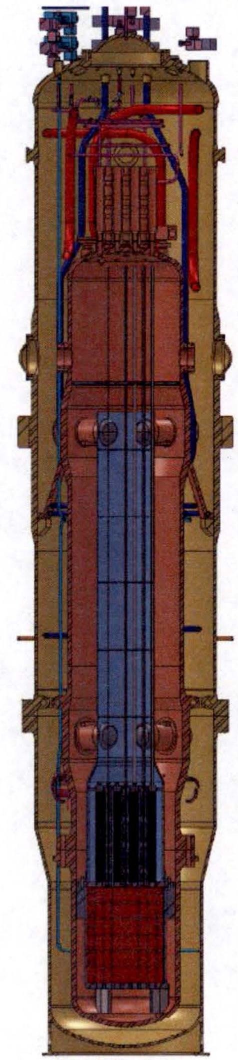
Power Module (RXM) Assembly


RPV



RPV


RXM

Three Power Module Components

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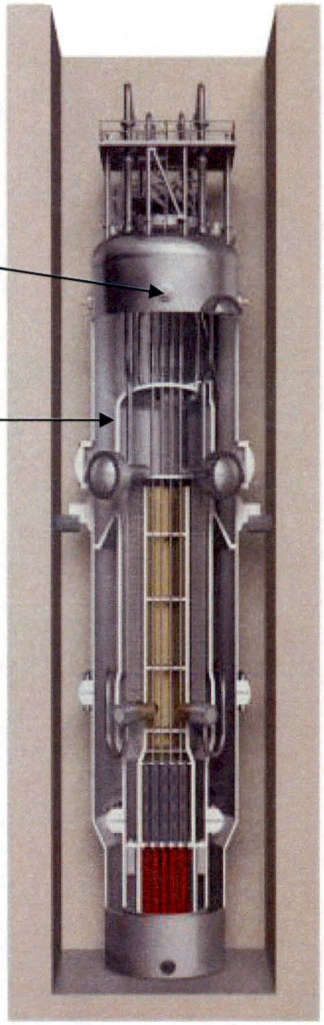
}}2(a),(c),ECI

Module Containment Design

Evacuated containment—enhanced safety

- Containment volume sized so that core does not uncover following a LOCA
- Large reactor pool keeps containment shell cool and promotes efficient post-LOCA steam condensation
- Insulating vacuum
  - significantly reduces conduction and convection heat transfer during normal operation
  - eliminates requirement for insulation on the reactor vessel, thereby minimizing sump screen blockage concerns (GSI-191)
  - improves LOCA steam condensation rates by eliminating air
  - Mitigates combustible hydrogen mixture in the unlikely event of a severe accident (i.e., little or no oxygen)
  - reduces corrosion and humidity problems inside containment



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Comparing PWR Containment Designs

Parameter	NuScale	AP1000	US-APWR
Containment Structure	**cylindrical/steel**	cylindrical/steel	PCCV cyl/steel liner
Containment Height (ft)	**75.8**	215	226.5
Containment Diameter (ft)	**15.0**	130	149
Length-to-Diameter Ratio	**5.06**	1.65	1.52
Material	{{ }}2(a),(c)	SA-738, Gr. B carbon steel	SA-516, Gr. 60 carbon steel
Yield Strength (ksi)	{{ }}2(a),(c)	60	32
Shell Thickness (in)	{{ }}2(a),(c)	1.75	0.25
{{ }}2(a),(c)	**yes**	no	no
Net Free Volume (ft^3)	{{ }}2(a),(c)	2,060,000	2,800,000
Internal Design Pressure (psig)	{{ }}2(a),(c)	59	68
Design Temperature (°F)	{{ }}2(a),(c)	300	300

PCCV = pre-stressed, post-tensioned concrete containment vessel

Note: Data for AP1000 and US-APWR obtained from DCD documentation.

[1] Clad on both sides by {{ }}2(a),(c)

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Module Seismic Support Configuration

- Vertically supported by CNV skirt
- CNV skirt horizontal restraints

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Reactor Pressure Vessel (RPV) Supports

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Module Three Dimensional (3D) model

- 3D ANSYS model of the RXM includes pool water effects

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Module 3D Model

The RXM model includes sub-models of RXB pool, CNV, RPV, lower RVI, upper RVI, CRDMs, supports, and fuel assemblies

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Helical Coil Steam Generator (HCSG)

HCSG Design

- Two HCSGs in the power module (each rated at 50% capacity)
- Physically integral within the upper power module component
- HCSG heat transfer surface area fits into a smaller height and volume than once through or U-tube steam generator design
- Properties of each HCSG

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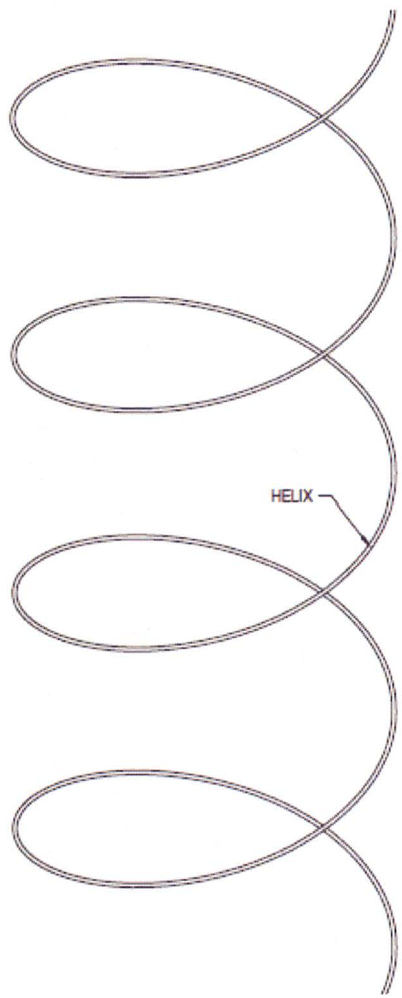
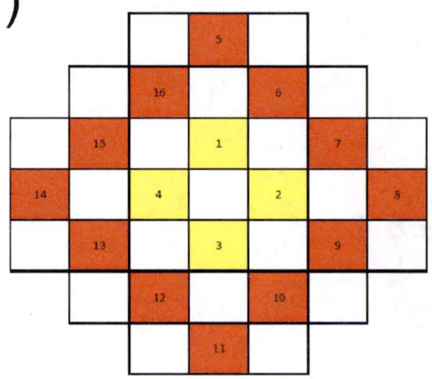
HCSG Single Tube and All Tubes

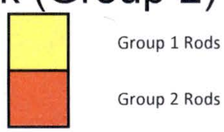


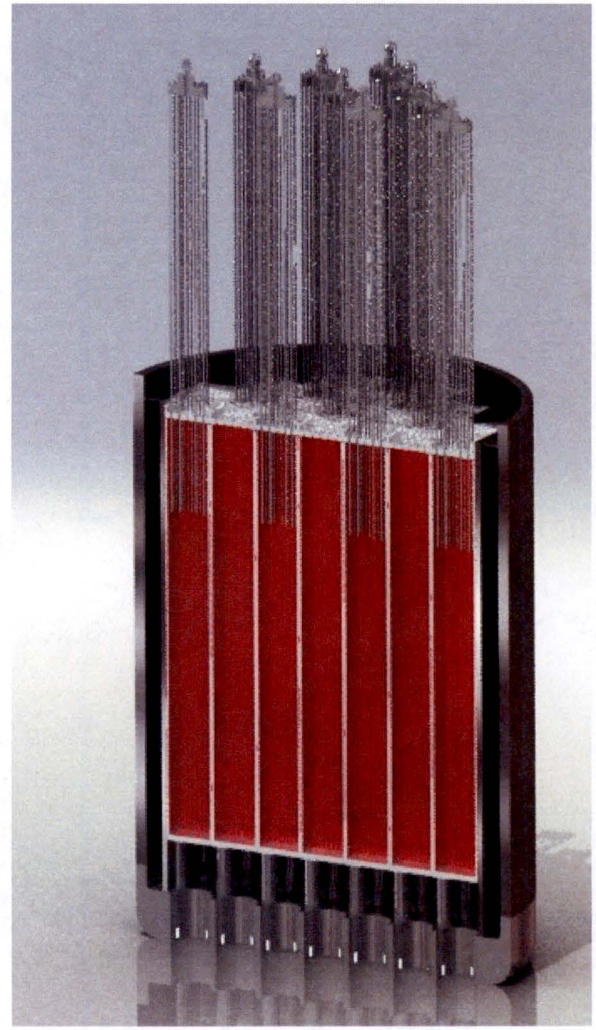
Core and Fuel Design

Reactor Core

- 17x17 lattice
- Approximately half-height
- 37 fuel assemblies
- UO_2 fuel pellets
- Clad material – AREVA M5® advanced cladding
- Negative reactivity coefficients
- 24 month cycle length at 95% capacity factor (695 effective full power day)
- U-235 enrichment < 4.95 %
- 16 control rod assemblies
  - 4 in regulating bank (Group 1)
  - 12 in shutdown bank (Group 2)







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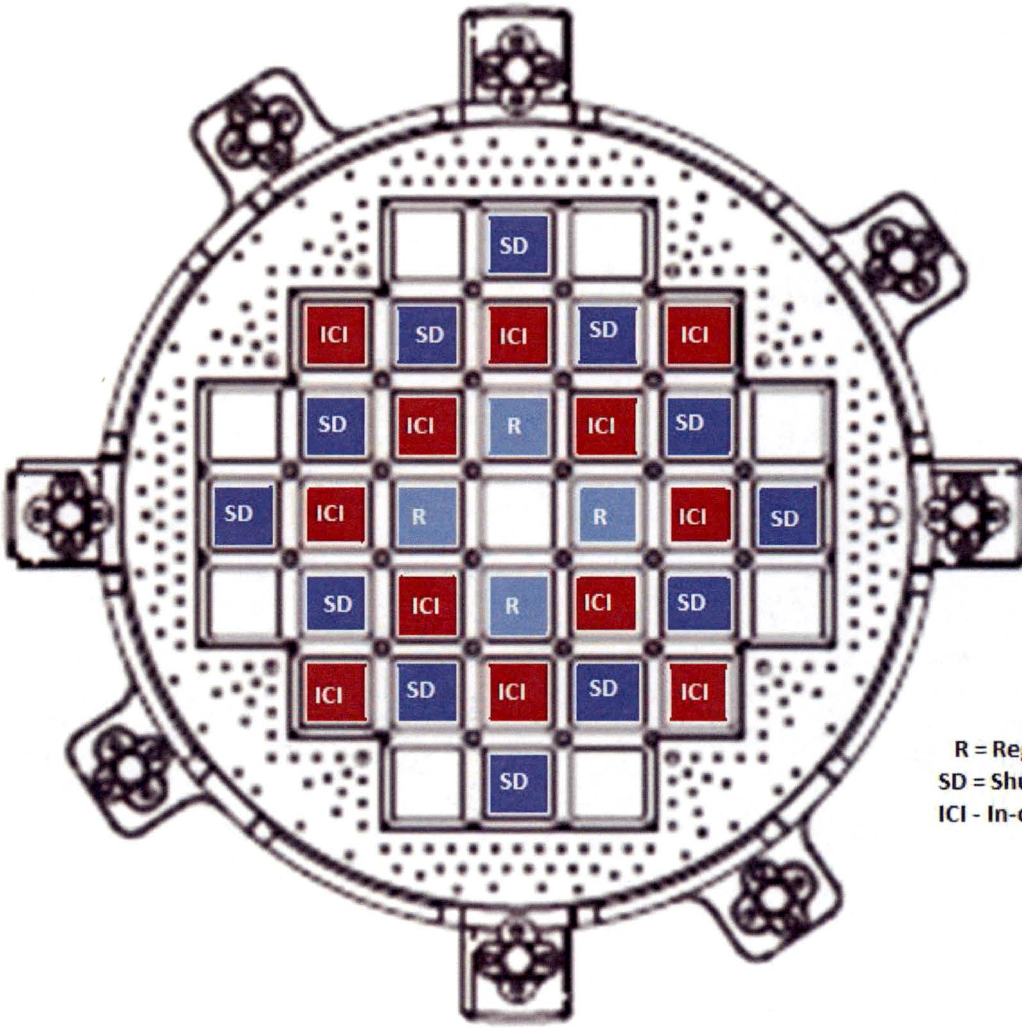
Core Configuration



Ex-core neutron detector

R = Regulating bank
SD = Shutdown bank
ICI - In-core instrument

Core Design: Basic Plant Parameters

Reactor Core Description	
• Thermal power rating	160 MW_{TH} (gross)
• Nominal operating pressure	12.7 MPa (1850 psia)
• Nominal core inlet, average, and exit temperature {{	}}2(a),(c)
• Number of assemblies	37
• Core weight {{	}}2(a),(c)
• Assembly pitch	21.504 cm
• Refueling interval	24 months {{ }}2(a),(c)
Fuel Assembly Description	
• Lattice geometry	17x17
• Enrichment	UO_2 (< 4.95% U^{235} enrichment)
• Fuel Rods per Assembly	264
• Guide/Instr. Tubes per Assembly	24/1
• Control Rod Material	Hybrid AIC and B4C
Fuel Rod Description	
• Active core height	2.0 m
• Burnable poison {{	}}2(a),(c)
• Clad material	M5®
• Fuel Pellet OD	0.8115 cm
• Clad ID	0.828 cm
• Clad OD	0.950 cm

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Power Dependent Core Flow

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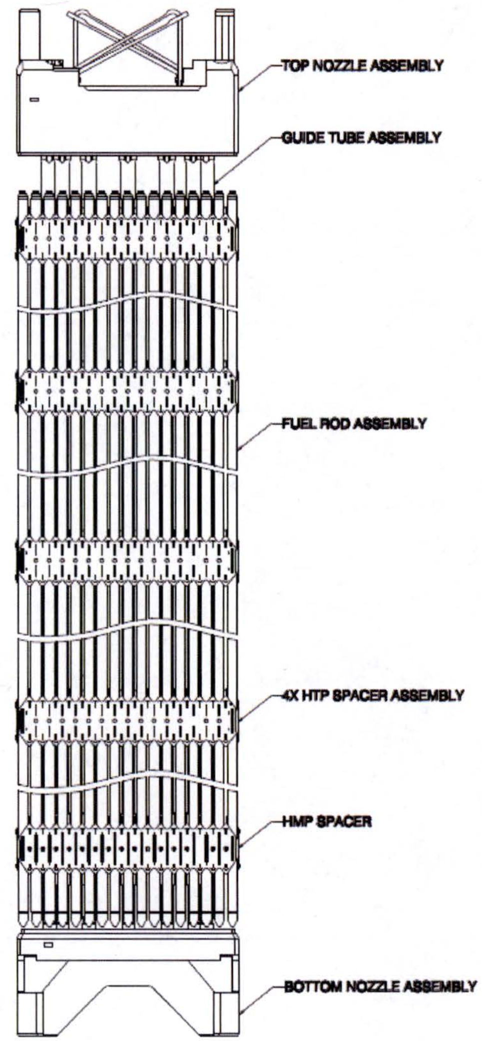
Percent core power }}2(a),(c)

Fuel Design – Proven Product

Component	Selection	Experience
Lower end fitting	Coarse Mesh Debris Filter	Standard grillage and filter plate; plant interface same as CE 16x16
Upper end fitting	QD attachment	AREVA standard product for 17x17 product line with hold down springs
Lower end grid	Inconel HMP	Identical to current AREVA production 17x17 product
Spacer grids	Zirc-4 HTP	Identical to current AREVA production 17x17 product
Fuel rod	M5™ Cladding 96% TD pellet	AREVA standard product; cladding and pellet dimensions identical to current 17x17 product
Guide tube	MONOBLOC Zirc-4	AREVA standard 17x17 product
Structure	Welded with bolted lower end fitting and QD upper end fitting	AREVA standard fabrication processes

NuScale Fuel Assembly Design

- NuScale design based on AREVA's proven US 17x17 PWR technology
- NuScale design features
  - Zircaloy-4 HTP™ upper and intermediate spacer grids
  - Inconel 718 HMP™ lower spacer grid
  - coarse-mesh filter plate on bottom nozzle
  - Zircaloy-4 MONOBLOC™ guide tubes
  - quick-disconnect top nozzle
  - Alloy M5® fuel rod cladding



>>*Proven features with significant US operating experience*

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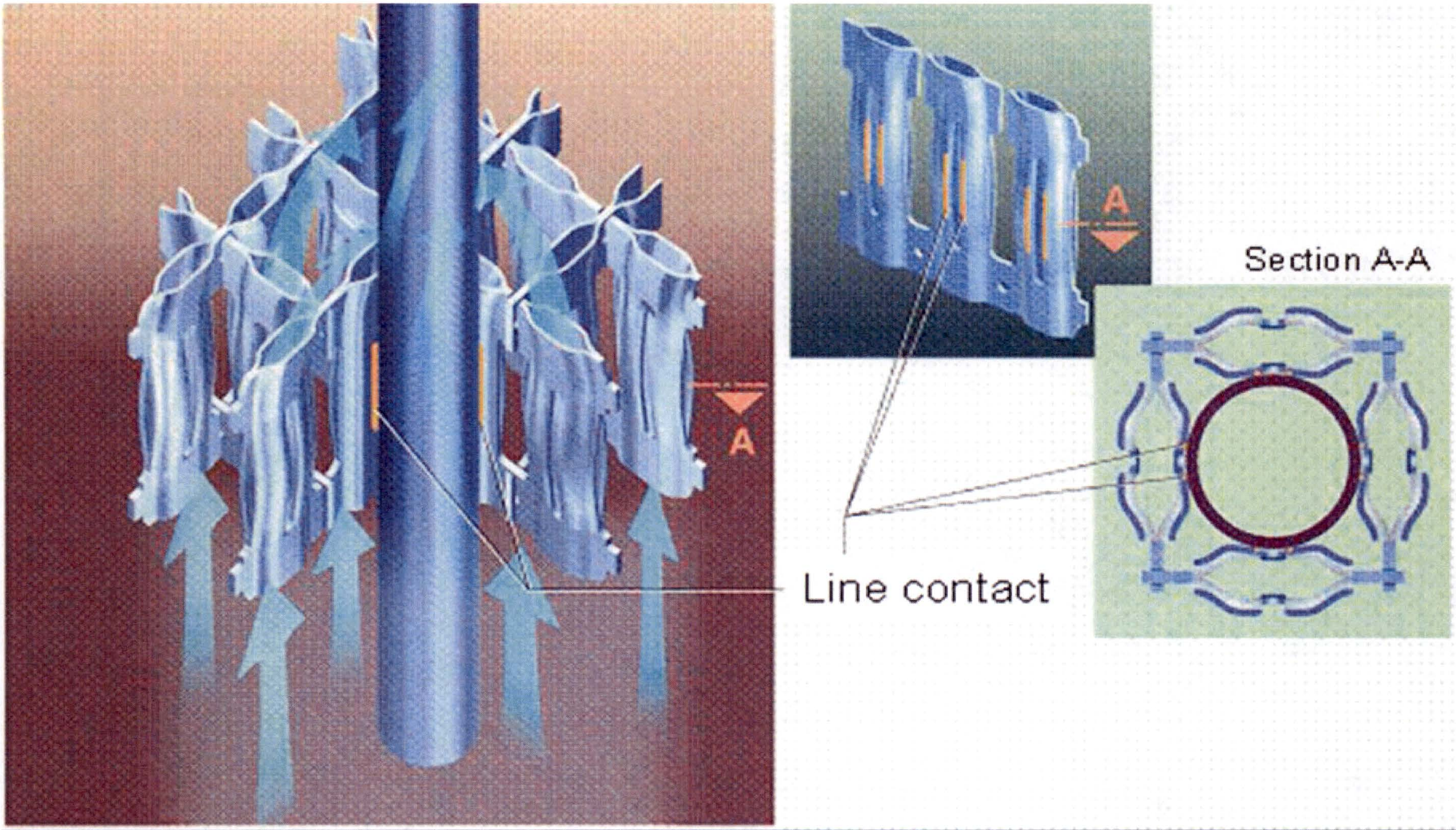
HTP Grid Design



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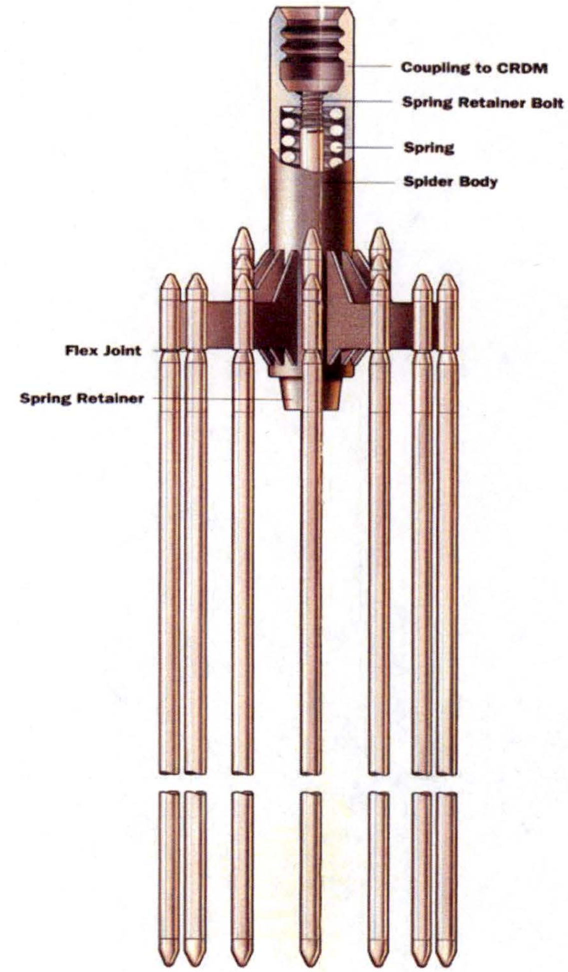
NuScale Control Rod Assembly Design

- Control rod assembly design based on AREVA's proven US 17x17 PWR technology
  - hybrid design – B_4C (boron carbide) and AIC (silver-indium cadmium) absorbers
  - 24 control rods with stainless steel cladding in each control rod assembly
  - one-piece cast stainless steel spider
  - flex joint formed by the combination of the pin, nut, upper end plug, and spider boss



>>Proven features with significant US operating experience

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Control Rod Drive System Supports

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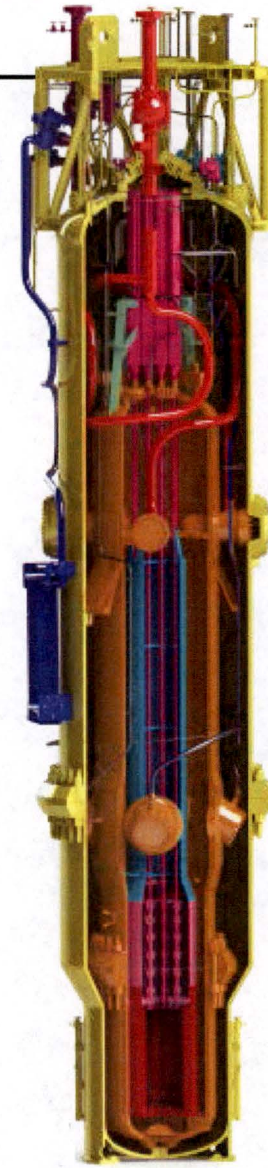

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Systems

Typical LWR Safety Systems

Systems and components needed to protect the core:

- Reactor pressure vessel
- Containment vessel
- Reactor coolant system
- Decay heat removal system
- Emergency core cooling system
- Reactor protection system
- Containment isolation system
- Ultimate heat sink
- Residual heat removal system
- Safety injection system
- Refueling water storage tank
- Condensate storage tank
- Auxiliary feedwater system
- Emergency service water system
- Hydrogen recombiner or ignition system
- Containment spray system
- Reactor coolant pumps
- Safety related electrical distribution systems
- Alternative off-site power
- Emergency diesel generators
- Safety related 1E battery system
- Anticipated transient without scram (ATWS) system

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NuScale Safety Systems

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- Safety related 1E battery system
- Anticipated transient without scram (ATWS) system

NuScale Power™

Design Simplification

- **New systems**
  - containment evacuation
  - containment flooding and drain
- **Eliminated systems**
  - containment spray
  - containment fan cooler
  - auxiliary feedwater
  - ECCS injection and recirculation
  - steam generator blowdown
  - electrical generator hydrogen supply
  - safety-related electrical systems
- **Eliminated components**
  - reactor coolant pumps
  - ECCS pumps, tanks, and RPV injection lines
  - containment sumps and tanks
  - refueling water storage tank
  - reactor coolant hot leg and cold leg piping
  - pressurizer surge line and relief tank
  - reactor vessel and primary coolant system insulation
  - safety-related emergency diesel generators

Containment Flood and Drain System (CFD)

- One system for six power modules (one on each side of the pool) connects to the module in its bay
- Fills CNV with borated UHS pool water from a connection above the minimum allowable power module pool level during shutdown and under certain beyond design basis accident conditions
- Removes water from the CNV in preparation for power operation after refueling
- Each system has two 100% capacity ({{ }}$^{2(a),(c)}$) centrifugal pumps that can fill the CNV up to the pressurizer baffle plate in less than {{ }}$^{2(a),(c)}$
- Piping, valves, and controls for pump suction from the UHS pool and discharge to the CNV through the containment evacuation system {{ }}$^{2(a),(c)}$ inside the CNV

Containment Evacuation System (CES)

- One separate and independent system for each power module
- Maintains low vacuum pressure ({{ }}2(a),(c)) in CNV during operation
- Remove and transfer water vapor and gases form the CNV; monitor water vapor and gases for radioactivity
- Provide RCS leak rate detection function and allow leak-before-break (LBB) methodology for FW and MS piping within the CNV
  - can detect 0.012 gpm RCS leak into CNV (meets RG 1.45)
  - detection response time < 1 hour for a 1 gpm leak (meets RG 1.45)
- Two 100% capacity vacuum pumps connected to a nozzle at the top of the CNV
- Vapor condenser and sample vessel
- Discharge connected to liquid and gaseous radwaste systems

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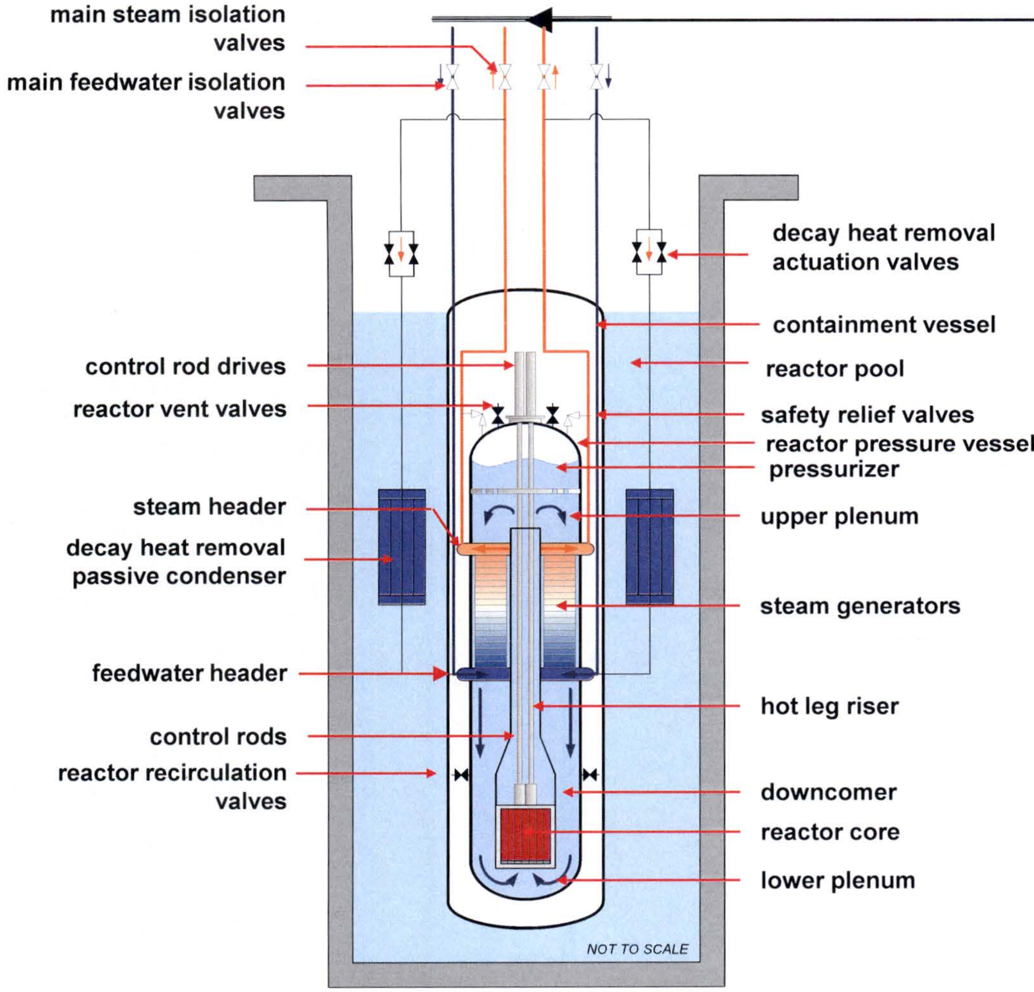
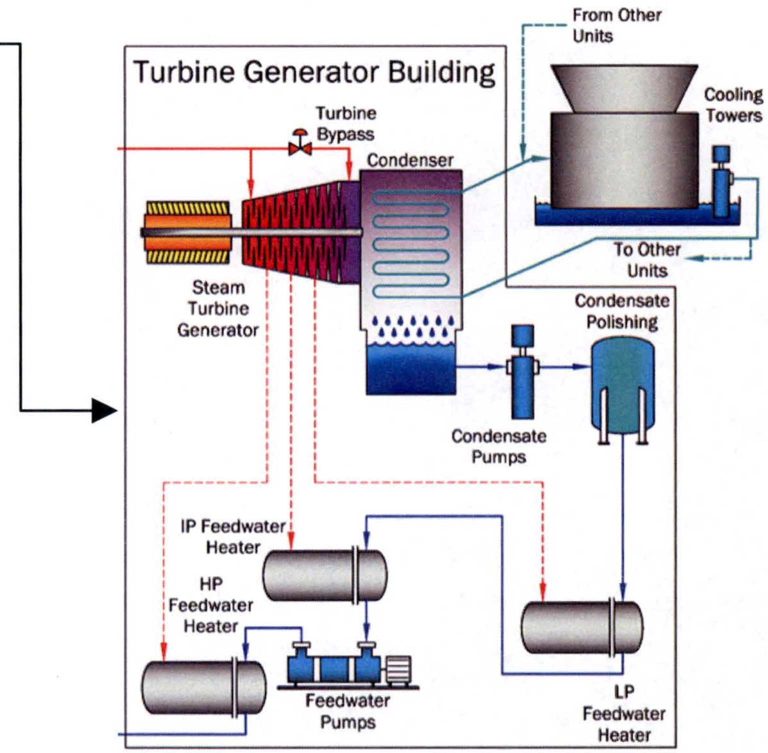
Some Auxiliary Systems

- Spent fuel pool cooling and cleanup system
- Light-load handling system (related to refueling)
- Overhead heavy load handling system
- Reactor component cooling water system
- Demineralized water system
- Potable and sanitary water systems
- Ultimate heat sink
- Condensate storage facilities
- Site cooling water system
- Chilled water system
- Compressed air system
- Process sampling system
- Equipment and floor drainage system
- Chemical and volume control system
- **Containment evacuation and flooding systems (unique to NuScale)**
- Reactor building and spent fuel pool area ventilation system
- Radwaste building ventilation
- Turbine building ventilation system

NuScale Power™

NuScale Power Train





- Each NuScale power module feeds one turbine generator (TG) train eliminating single-shaft risk
- 100% turbine bypass capability
- Small, simple components support short, simple refueling outages

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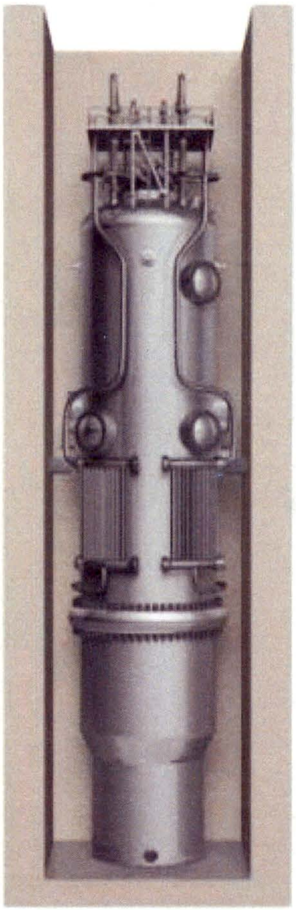
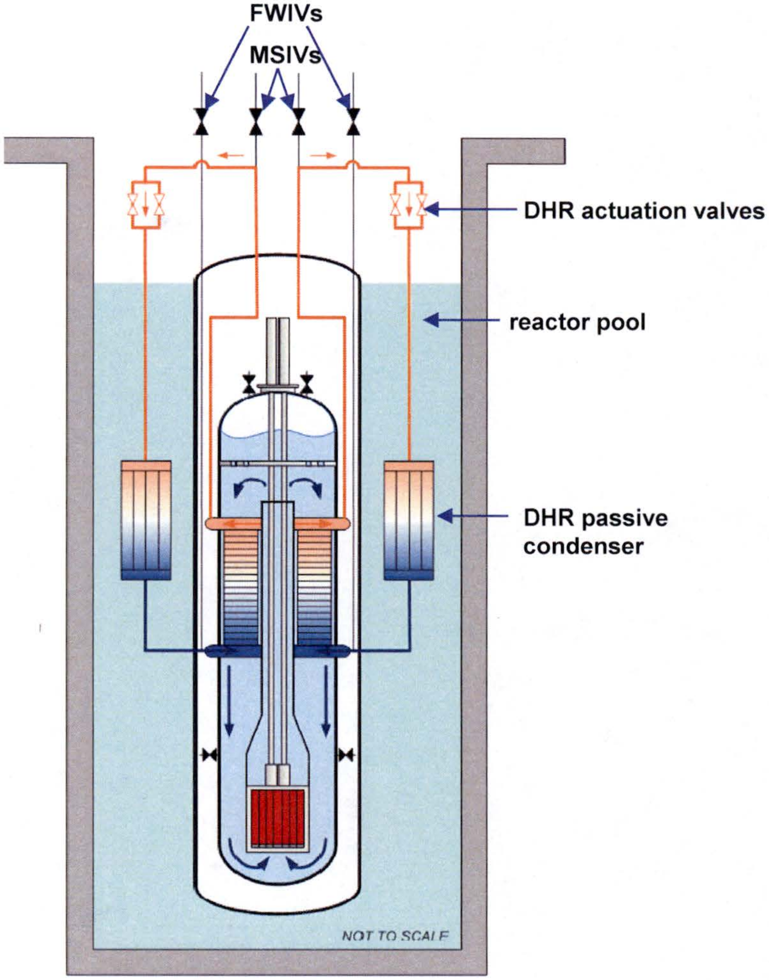
Decay Heat Removal System

- Main steam and main feedwater isolated
- Decay heat removal (DHR) valves opened
- Decay heat passively removed via the steam generators and DHR heat condensers to the reactor pool
- DHR system is composed of two independent single failure proof trains (1 of 2 trains needed)





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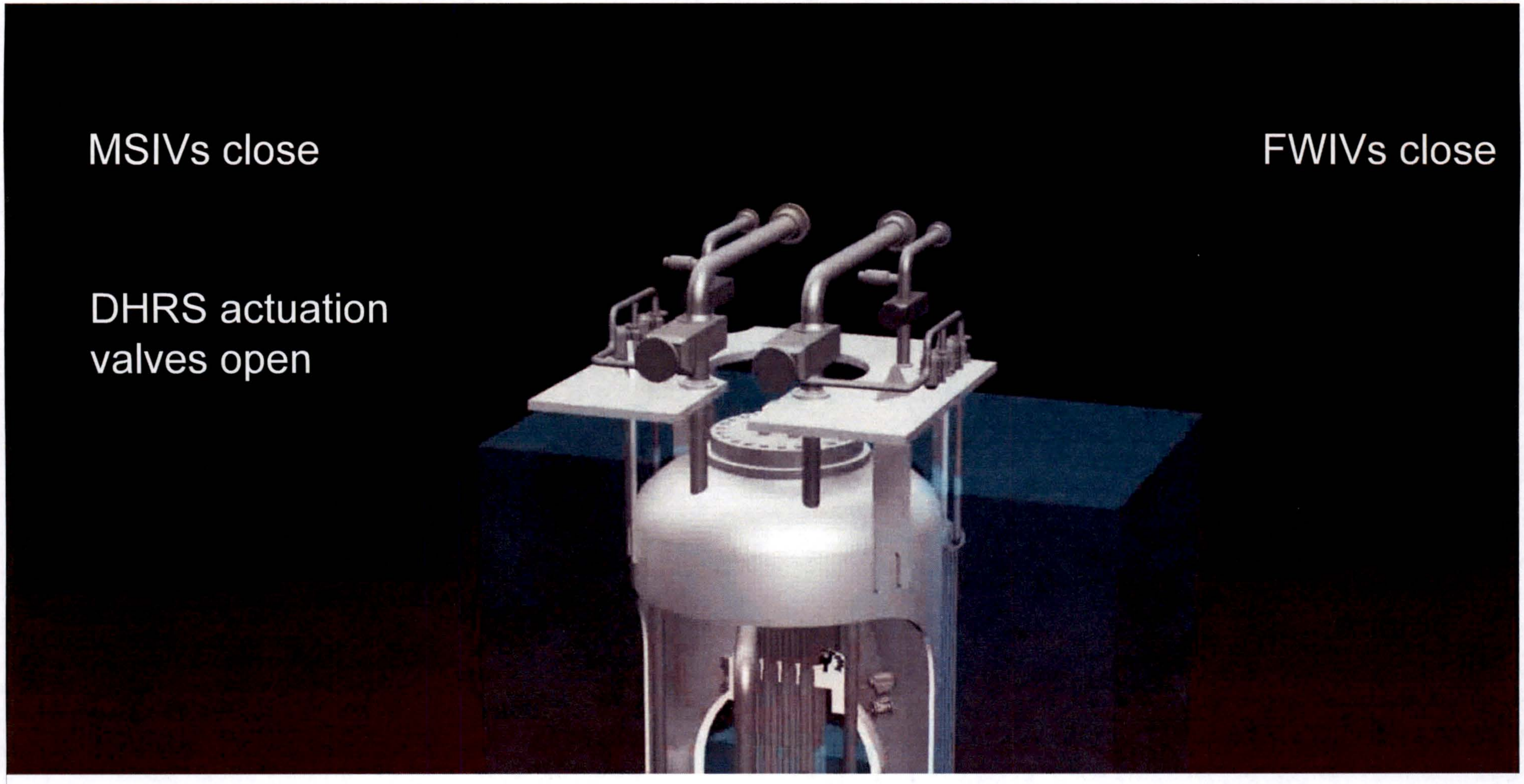
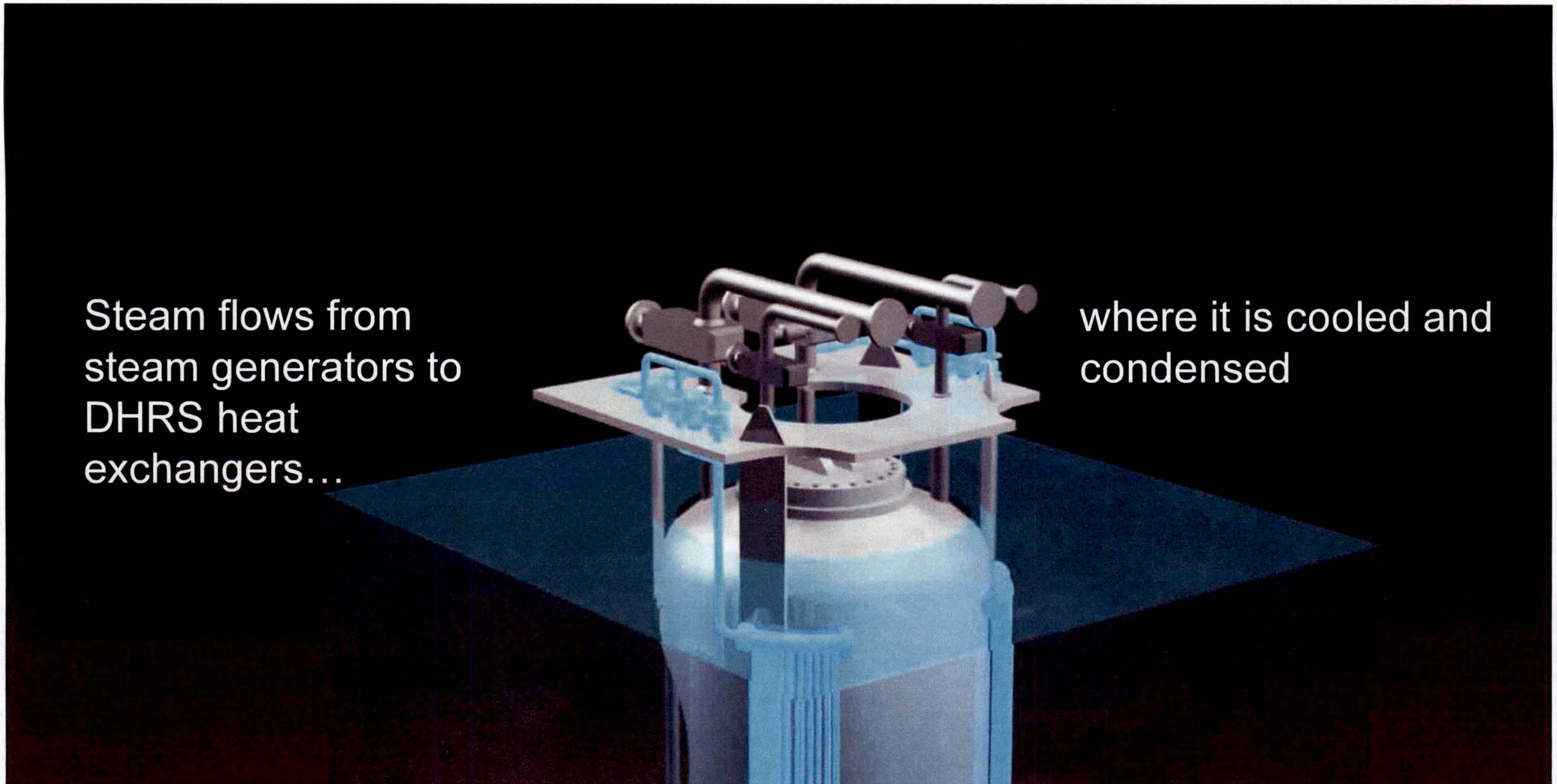
DHRS Operation



DHRS Operation



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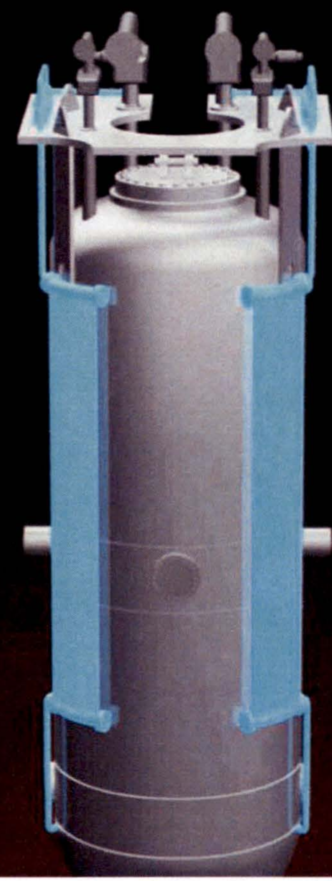
DHRS Operation

Water gravity drains back into steam generator tubes



Reactor coolant heat is transferred through tubes

Water inside tubes once again becomes steam

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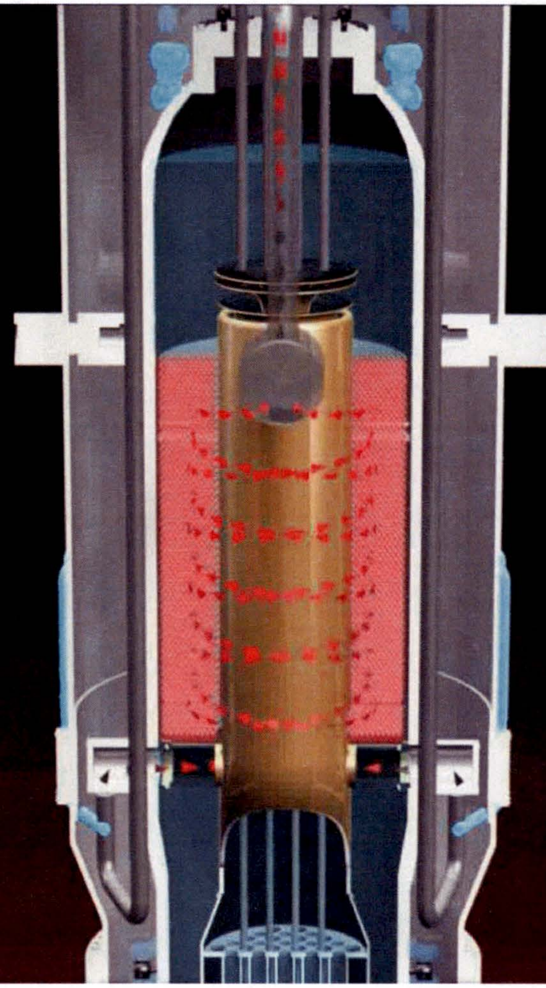
DHRS Operation

Steam flows out of steam generators…

and back to DHRS heat exchangers

Heat is removed by the UHS



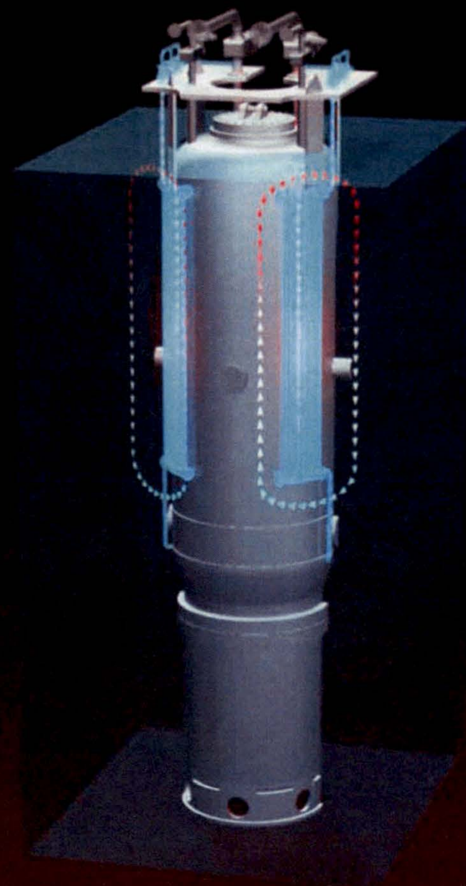
Closed loop, natural circulation flow is established in both DHRS trains

PM-0616-49510-NP
Revision: 0

DHRS Operation

Heat is transferred

- Core to coolant
- Coolant to DHRS through the steam generator tubes
- DHRS to UHS
- No electrical power needed
- No operator actions needed



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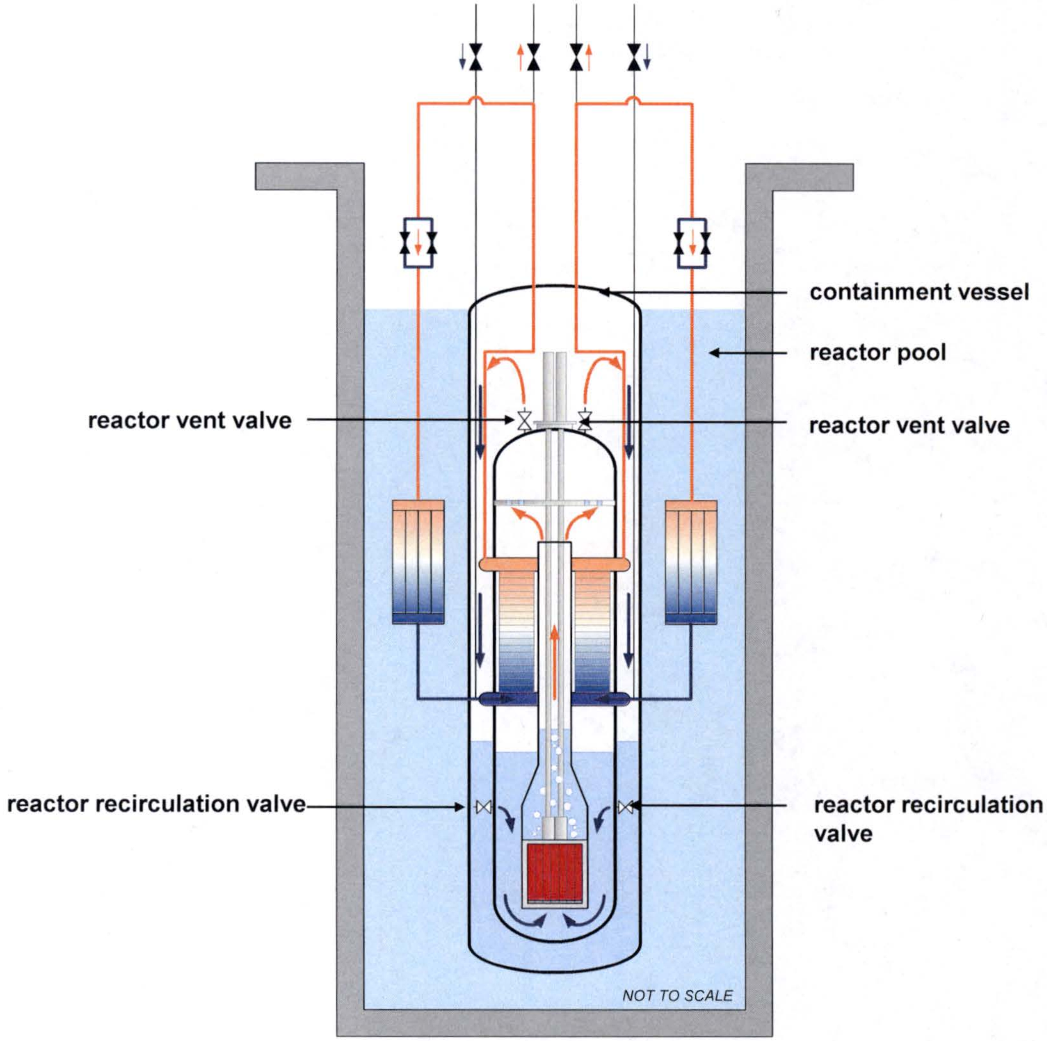
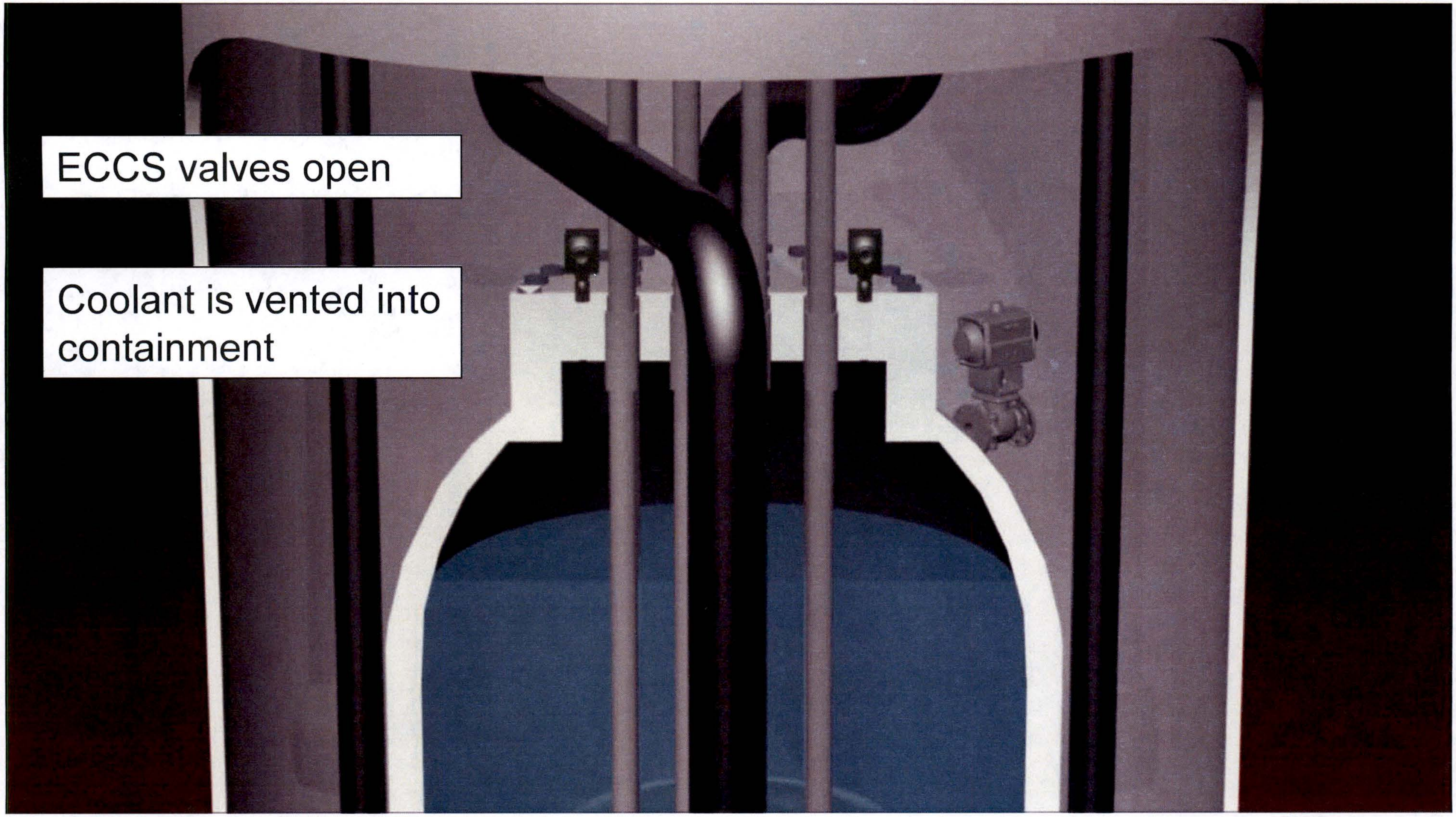
Revision: 0

Emergency Core Cooling System

- Main steam and main feedwater isolated
- Reactor vent valves and reactor recirculation valves open on safety signal
- Decay heat removed
  - condensing steam on inside surface of containment vessel
  - convection and conduction through liquid and both vessel walls



NuScale's ECCS Operation



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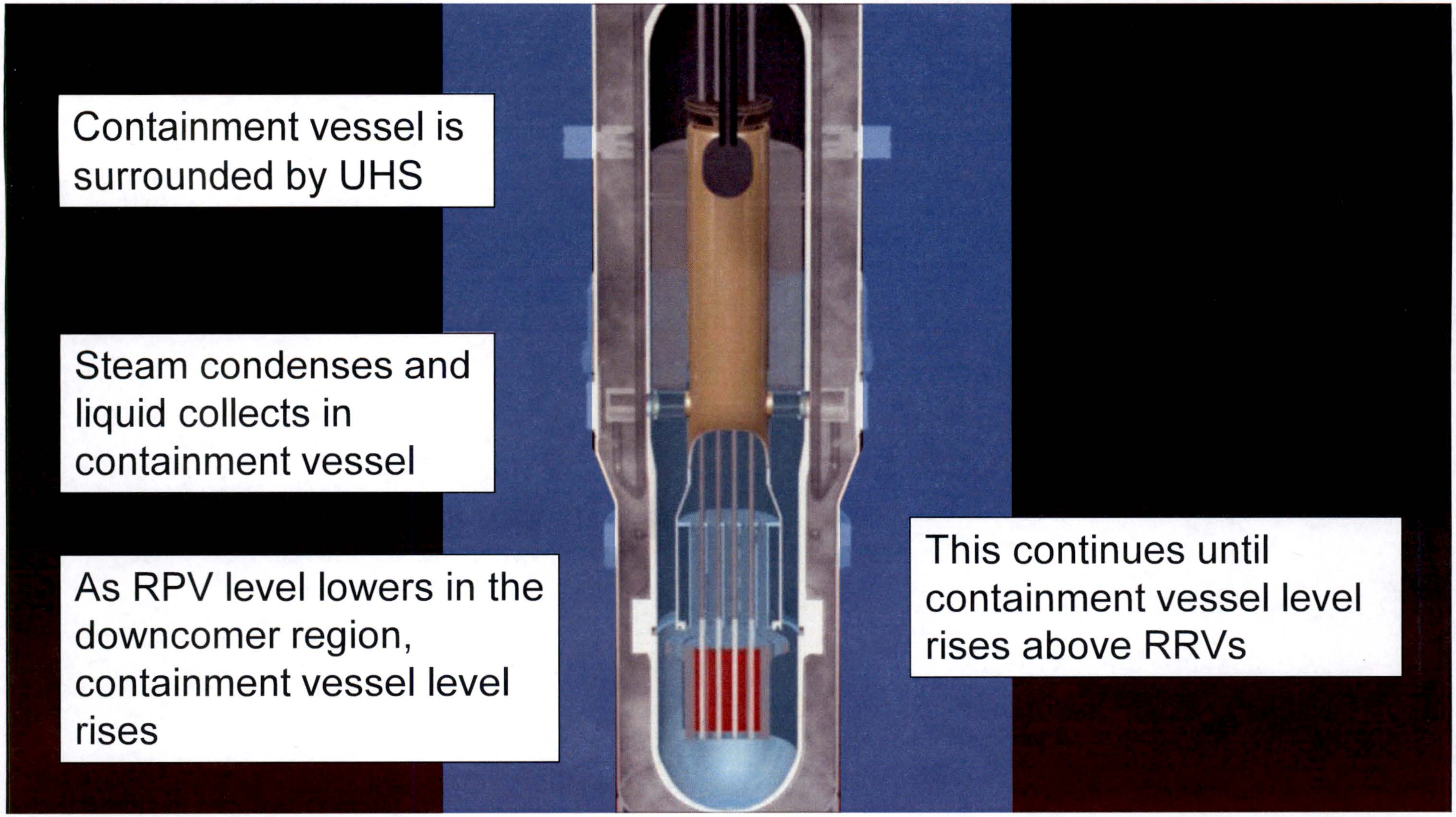
NuScale's ECCS Operation



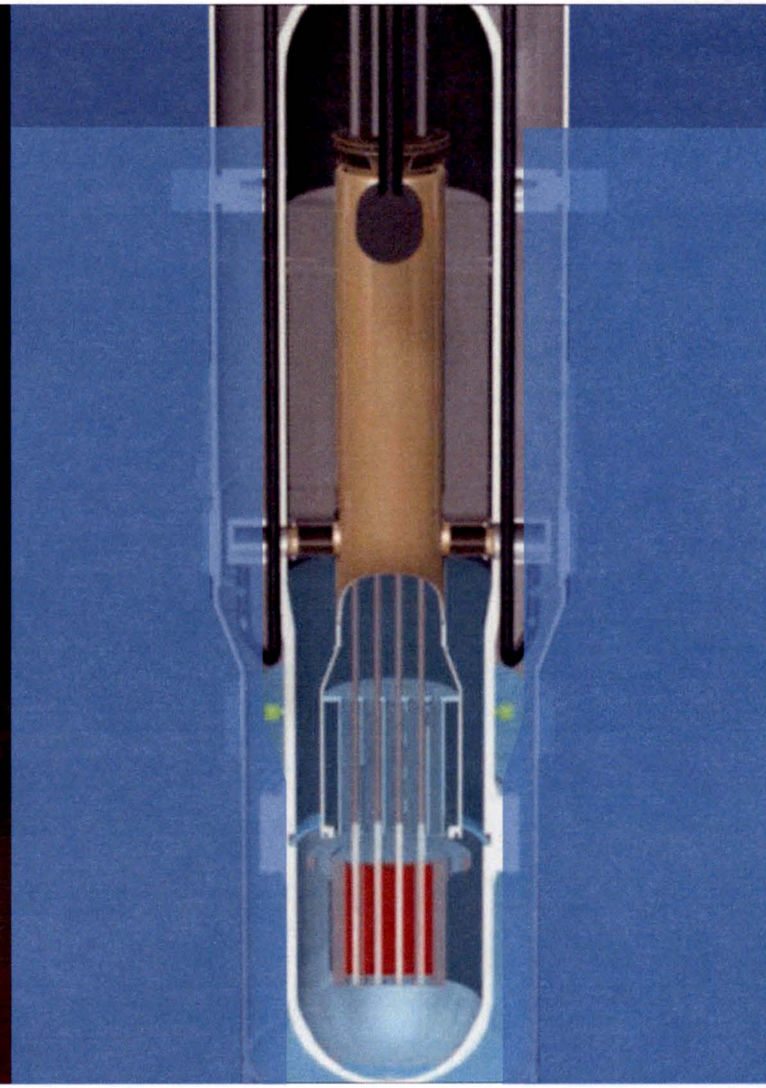
NuScale's ECCS Operation

Coolant flows from the containment vessel into the RPV

Flows to bottom of RPV and back to the core

Heated by the core

Flows up to top of RPV



Coolant again exits through RVVs

Heat is transferred
- Core to coolant

Repeats the cycle

containment vessel
- Containment vessel to UHS
- No electrical power needed
- No operator actions needed

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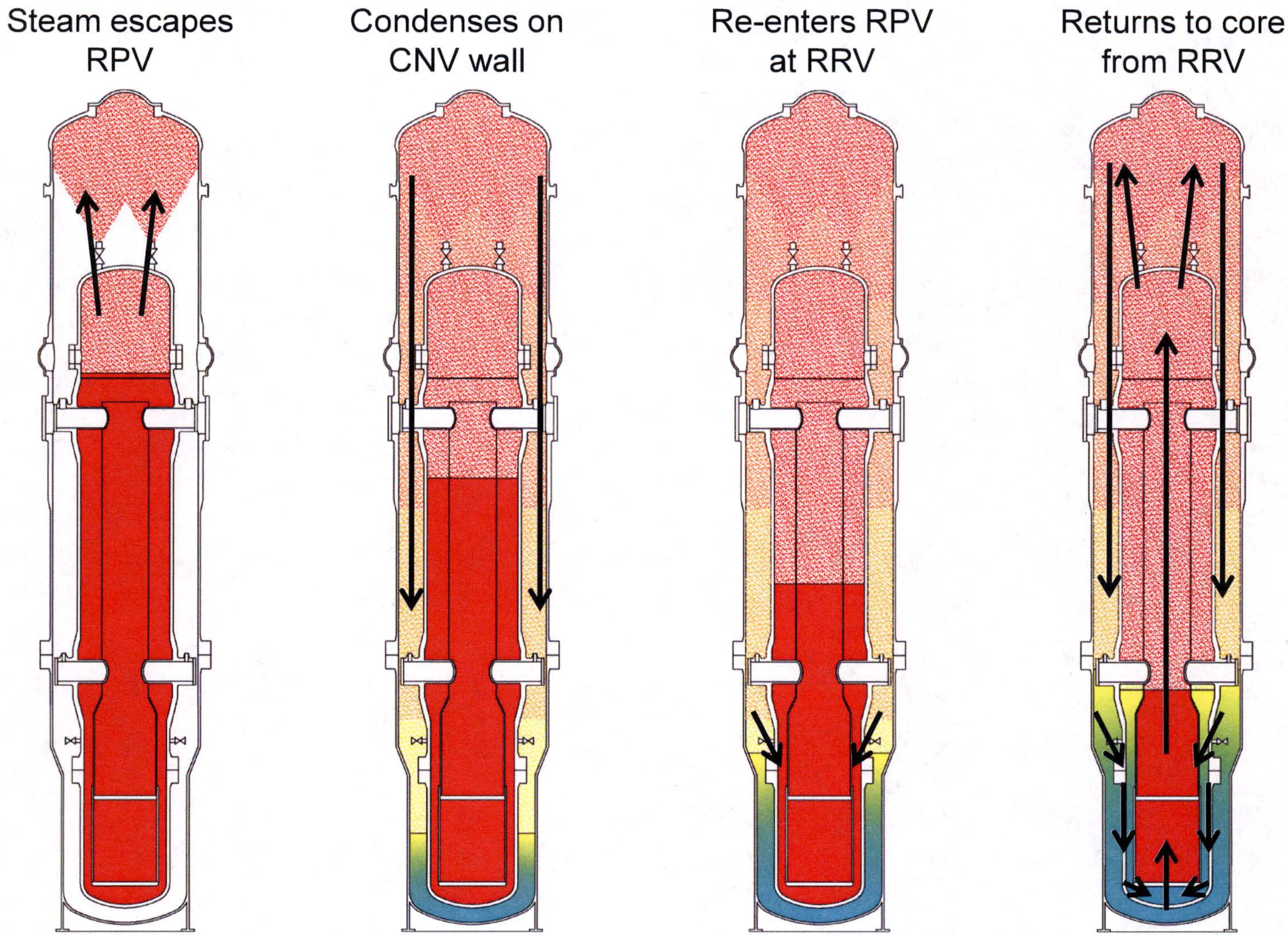
Accident Operation



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ECCS Valve Design

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}}2(a),(c)

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ECCS Actuation and Operation

- ECCS actuates on any of the following:

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}}2(a),(c)

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Chemical Volume and Control System

- **Maintain RCS pressure (pressurizer spray)**
- Maintain RCS inventory (pressurizer level)
- Control soluble boron concentration in RCS
- Remove radionuclides from the primary coolant
- Chemistry control (LiOH, hydrazine, zinc injection) of RCS
- **Heatup RCS during startup (reactor startup heater)**
- **Develop natural circulation flow during startup**
- Pressurize the pressurizer with a nitrogen gas bubble during startup
- Vent non-condensable gases from the primary coolant pressurizer
- Extract RCS samples for chemical and radionuclide analysis during normal and post-accident conditions

Functions in **bold** are unique to the NuScale CVCS and performed by other systems in large PWRs

CVCS Components

- Module dedicated
  - charging and recirculation pumps ({{ }}2(a),(c))
  - regenerative and non-regenerative heat exchangers
  - chemical mixing tank
  - letdown, charging, and pressurizer spray lines
  - ion exchangers {{ }}2(a),(c)
  - RCS filters with {{ }}2(a),(c)
  - expansion tank
- Common to multiple modules
  - 2- 50% capacity module heatup heat exchangers for each 6 modules
  - boric acid batch tank, transfer pump, storage tank, and supply pumps
  - nitrogen supply system

Power Module Piping and Valves

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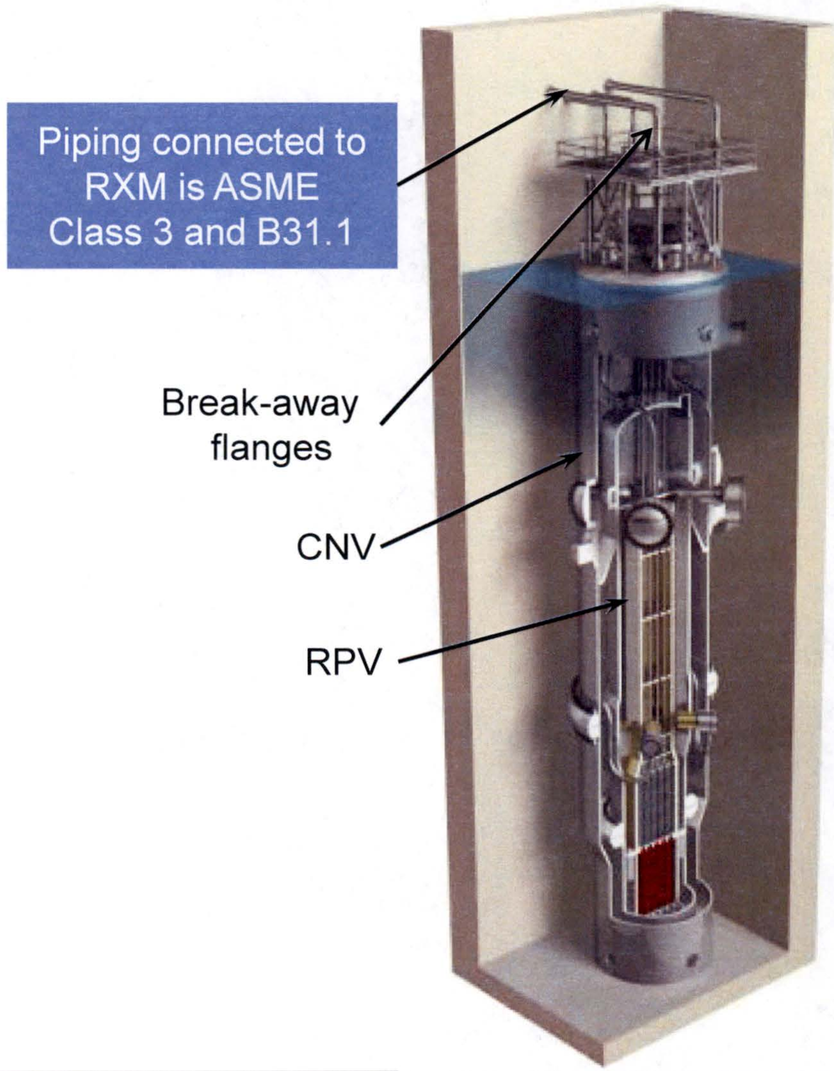
Power Module (RXM) Piping Systems

- The RXM includes ASME Class 1, 2, and 3 piping
- Piping inside containment is ASME Class 1 and 2 and is uninsulated
- RXM is fabricated in the factory along with all piping up to breakaway flange



Piping System Design

{{

}}2(a),(c),ECI

CVCS=chemical volume and control system; MS=main steam; FW=feedwater; DHRS=decay heat removal system; RCCW=reactor component cooling water; CFDS=containment flood and drain system; CES=containment evacuation system

{{

}}2(a),(c),ECI

Module Piping ASME Class Boundaries

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}}2(a),(c),ECI

Power Module Piping Layout

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}}2(a),(c)

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Power Module Piping Layout

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}}2(a),(c)

Power Module Piping Layout

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}}2(a),(c)

NuScale Power™

CVCS Layout

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}}2(a),(c)

NuScale Power™

CVCS Layout

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}}2(a),(c)

Decay Heat Removal Layout

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}}2(a),(c)

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Feedwater Layout

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}}2(a),(c)

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Main Steam Layout

{{

}}2(a),(c)

LBB - Leak Detection Capability

- Containment vessel sealed and maintained at a vacuum low enough to avoid condensation
- The objective of the leak monitoring system is to detect:
  - leaks as small as 0.05 GPM (RG 1.45; SRP 5.2.5)
  - leaks of 1 GPM within 1 hour (RG 1.45; SRP 5.2.5)
  - leaks within LBB criteria (SRP 3.6.3)
- All leakage into CNV is commingled and unquantified until characterized (e.g., samples)

{{

}}2(a),(c)

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Revision: 0

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LBB - Leak Monitoring System

- **Two sensitive means to detect leaks**

  During startup, the system measures actual baseline leak rate and monitors for an increase

{{

}}2(a),(c)

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Principal Power Module Valves

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}}2(a),(c),ECI

Electric Power System

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NuScale Electrical Systems

- Normal power source: main generators
  - up to 12 generators with island mode capability
  - transmission grid is a load (does not supply power to the site)
- Backup power sources
  - Two DC systems (highly reliable and normal)
  - Two Backup Diesel Generators (BDGs)
  - Auxiliary AC Power Supply (AAPS) (e.g., combustion turbine generator)
  - Extended Loss of AC Power (ELAP) connections
- With loss of power
  - **electrical power not necessary to perform any safety function**
    - for safety functions, NuScale system response is the same whether actuated by actuation signal from MPS, manual actuation, or loss of power
  - important loads can be powered by DC, BDGs, and AAPS
  - Highly reliable DC powers post-accident monitoring instrumentation along with other loads
  - Other loads can be powered by AAPS

Highly Reliable Electrical System Design

- NuScale highly reliable DC power system (EDSS)

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}}2(a),(c)ECI

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Highly Reliable Electrical System Design

- NuScale highly reliable DC power system (EDSS)

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}}2(a),(c),ECI

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Highly Reliable Electrical System Design

- NuScale highly reliable DC power system (EDSS)

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}}2(a),(c),ECI

NuScale Power™

Highly Reliable Electrical System Design

- NuScale highly reliable DC power system

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}}2(a),(c),ECI

Instrumentation and Actuation Signals

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Power Module Instrumentation

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}}2(a),(c),ECI

Power Module and Pool Instrumentation (2)

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}}2(a),(c),ECI

NuScale Power™

Signals That Initiate Reactor Trip

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}}2(a),(c),ECI

Signals That Initiate DHRS/ECCS

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}}2(a),(c),ECI

Signals That Initiate CNV Isolation

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}}2(a),(c),ECI

Refueling

Initial Module Installation

- Module arrives in three parts:
  - upper module (includes both upper CNV and RPV)
  - lower CNV
  - lower RPV
- Module has factory ITAAC completed
- Lower CNV and lower RPV are placed in their respective tools and the upper module is placed in the import trolley, and then the following windows are completed:
  - initial fuel load
  - assembly
  - connection
  - module heatup
  - start-up

Refueling Overview

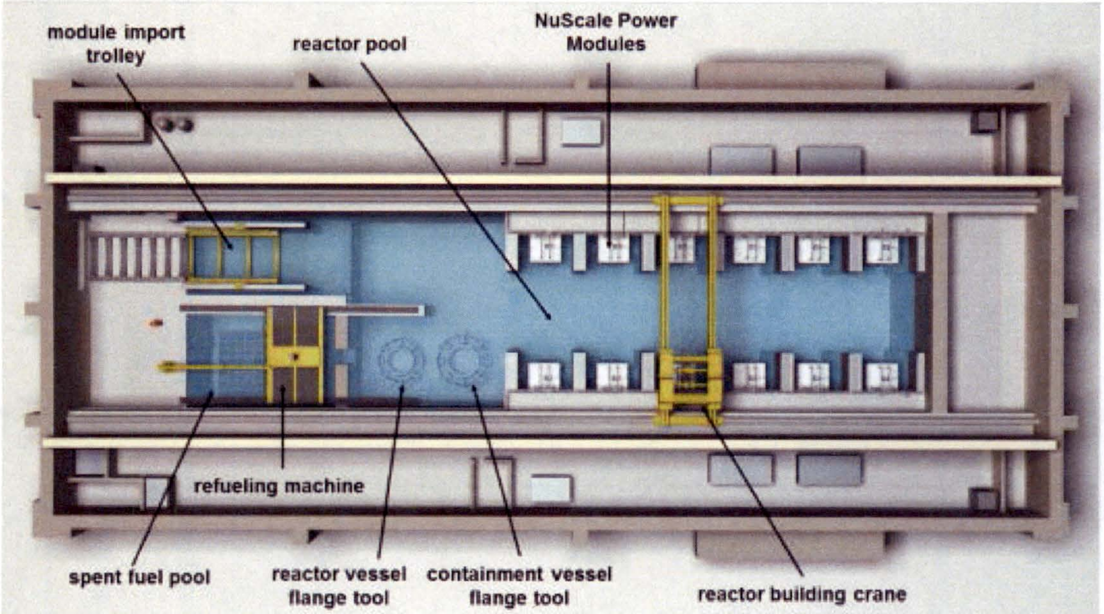
- Two-year refueling cycle for each module
- Dedicated refueling crew separate from the operations crew—including a dedicated refueling SRO
- Once a module is disconnected, the refueling crew assumes primary responsibility

NuScale Plant Refueling

Refueling operations

- Removal of connections from the operating bay
- Transport of the module to the refueling area
- Flange de-tensioning
- Module inspection
- Module is refueled
- Outage occurs every two years
- Refueling outage lasts 10 days from power down to power up

NuScale Plant Refueling

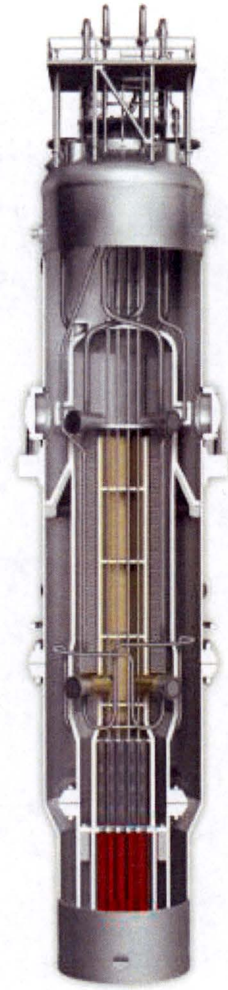


Windows

1. Shutdown/cooldown
2. Transition preparation and disconnection
3. Transition (to containment flange tool)
4. Disassembly
5. Upper module work window
6. Refueling
7. Lower containment vessel work window
8. Reassembly
9. Transition (to operating bay)
10. Reconnection
11. Module heatup
12. Reactor start-up and ramp to full power

Shutdown/Cooldown

{{



}}2(a),(c),ECI

Transition Preparation and Disconnection

- Shutdown CVCS
- Fully depressurize RCS
- Open ECCS valves
- Remove bioshield
- Close all containment isolation valves
- Electrical and I&C disconnections performed
- Containment is pressurized to prevent water from coming over the top of the RPV head when the CNV flange is separated
- Mechanical disconnections completed
- Crane and lifting device connected

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Simplified Graphic of Module During Transition

{{

}}2(a),(c)

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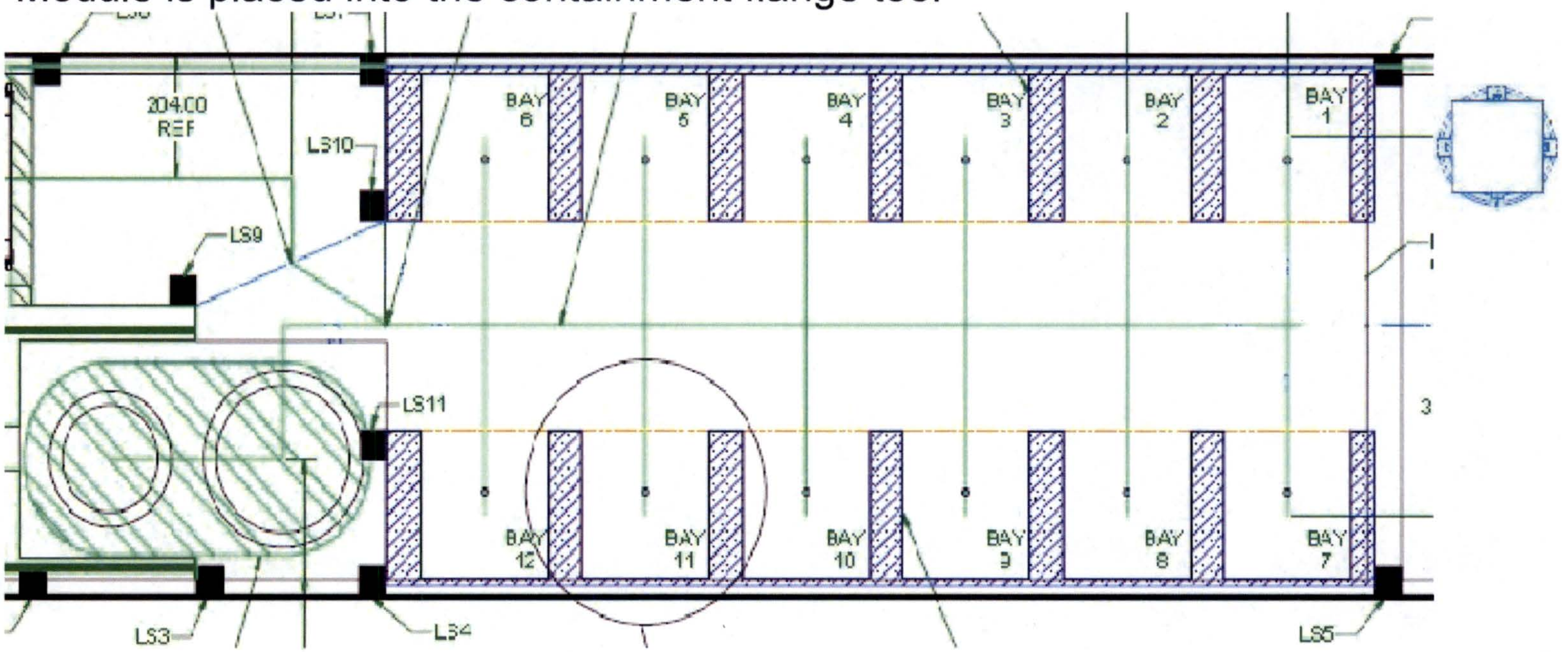
Module Transport

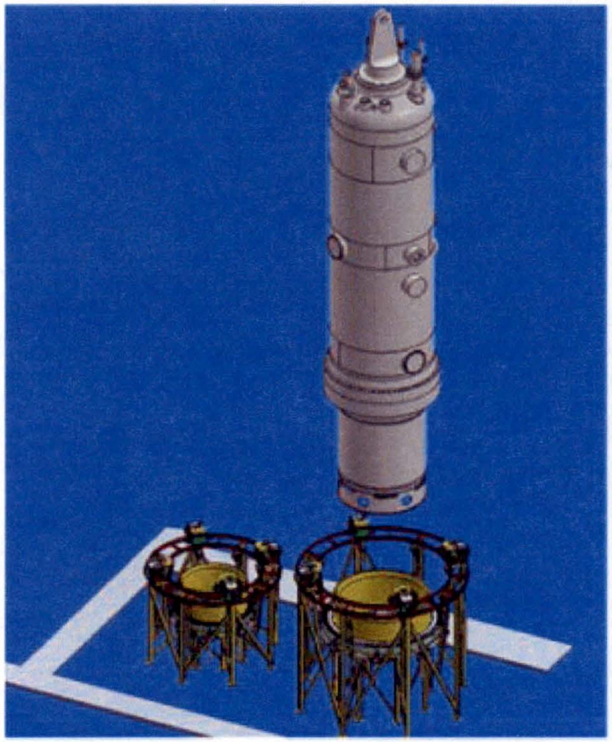
1. Module is raised enough to clear the pool floor support ({{ }}$^{2(a),(c)}$)
2. Crane trolleys to the centerline of the pool
3. Crane travels to the lift point (just past bay 6 and 12)
4. Module is briefly lifted {{ }}$^{2(a),(c)}$ above the UHS pool floor to place it into the containment flange tool
5. Crane travels along the centerline until it is lined up with the containment flange tool
6. Crane trolleys until the module is directly over the containment flange tool
7. Module is placed into the containment flange tool

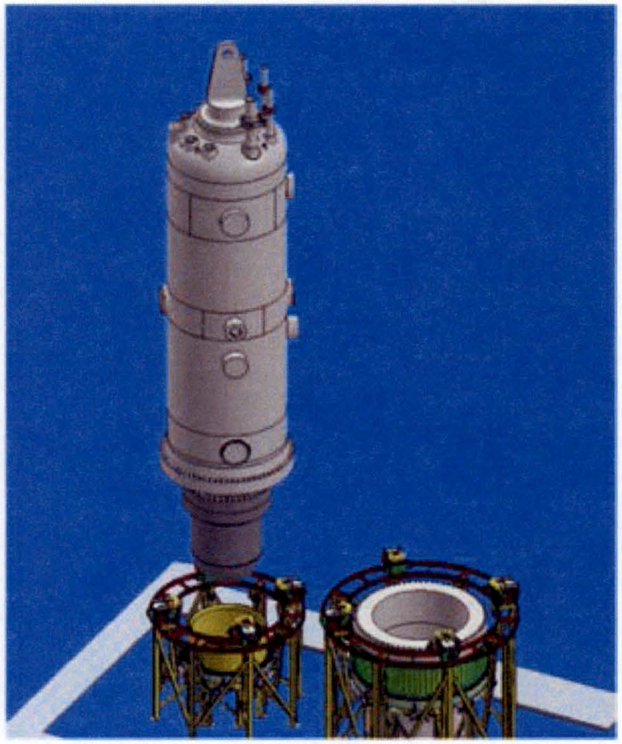


Disassembly

The containment flange tool detensions the containment flange studs, (refueling neutron monitoring instrumentation is put into place), and provides a stand for the containment lower vessel

The remainder of the module is picked up and moved to the reactor flange tool—where the reactor flange studs are detensioned, and then the upper module is lifted and transported to the dry dock, leaving the lower reactor vessel and the core for refueling





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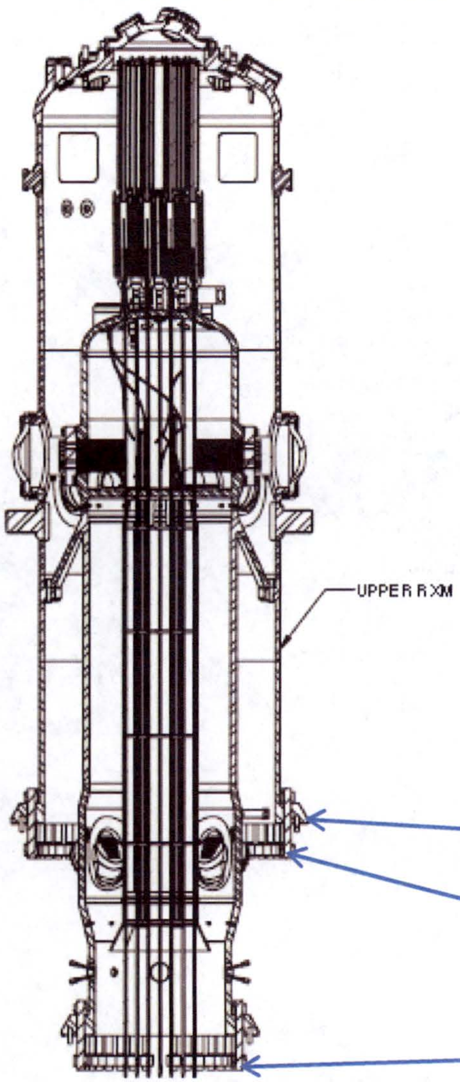
Upper Module Work Window



- Upper module is secured in the module inspection rack and the crane and lifting rig is removed
- Steam generator inspections
- Instrument testing, repair, and calibration
- Upper reactor flange inspection
- Upper containment flange inspection
- Perform ISI of ASME welds, forgings, and surfaces
- Perform IST of ASME valves (SRVs, RRVs, RVVs and check valves). There are no ASME pumps or dynamic restraints
- Appendix J Type B and C testing

steam generator feedwater plenum

upper containment flange

upper reactor flange

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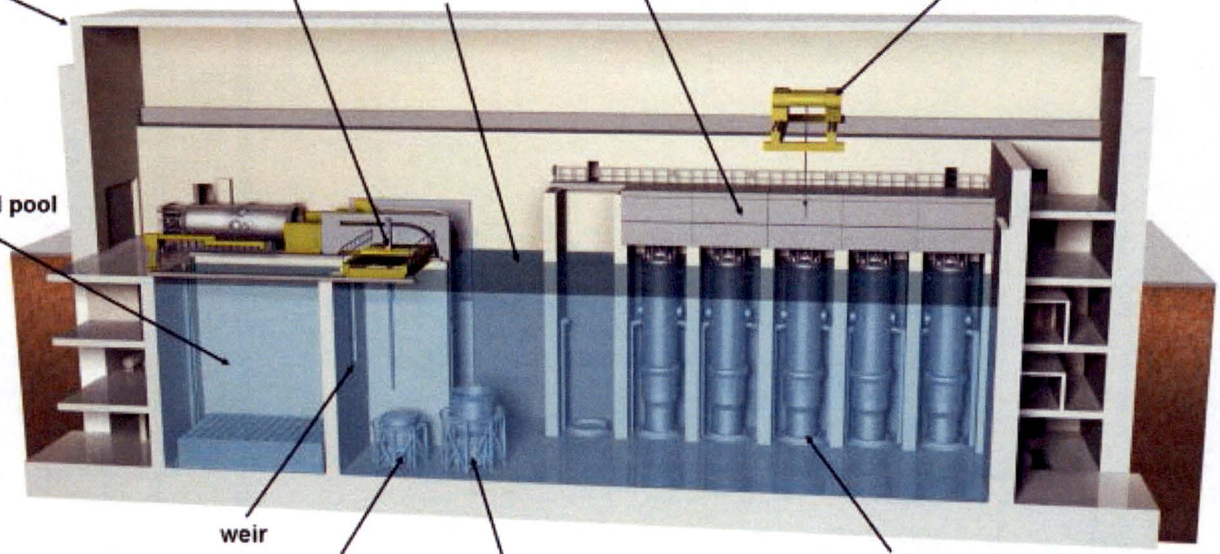
Refueling



The 37 fuel assemblies in the lower reactor vessel can be taken directly from the core with the refueling machine and placed in the spent fuel pool.

Each assembly only requires a single handling event to take it from the core to the spent fuel storage location.

Lower reactor vessel inspections would also be performed in this window.

Refueling can be completed either as a partial core offload and fuel shuffle, or full-core offload and reload.

Module Reassembly

- Prerequisites
  - lower containment work window complete
  - upper module work window complete
  - refueling work window complete
- Place upper module on lower RPV and tension flange
- Connect control rods and perform latch and stroke test
- Leak test reactor flange
- Place upper module on lower CNV and tension flange
- Leak test containment flange
- The module is then moved to the operating bay following the load path

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Module Reconnection

- Prerequisites
  - CVCS operating on recirculation
  - feedwater and condensate on long cleanup
- Connect power, instrumentation and controls
- Verify instrumentation and control operability
- Place ex-core Nuclear Instruments in their operating position
- Insert in-core instrumentation
- Connect CES and begin containment and RCS degas
- Complete the remainder of the mechanical connections
- Close and reset ECCS vent and recirculation valves
- Pressurize the RCS with nitrogen to provide NPSH for CVCS recirculation pumps

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Module Heatup

- Establish normal CVCS recirculation
- Place module heatup system in service
- Restore feedwater and main steam and perform SG flush
- Verify containment is operable and drain containment
- Install bioshield
- Perform first dilution towards critical boron concentration
- Draw a vacuum on containment
- Draw a steam bubble in the pressurizer
- Complete dilution to critical boron concentration
- Stabilize RCS temperature at {{ }}$^{2(a),(c)}$
- Stabilize RCS pressure at {{ }}$^{2(a),(c)}$

Module Start-Up

- Withdraw rods to criticality
- Perform physics testing
- Withdraw rods to raise power to {{ }}2(a),(c) and Tave to {{ }}2(a),(c)
- Remove the CVCS heater from service
- Place the main turbine in service
- Synchronize turbine generator to the grid
- Ascend to 100% power

Operations

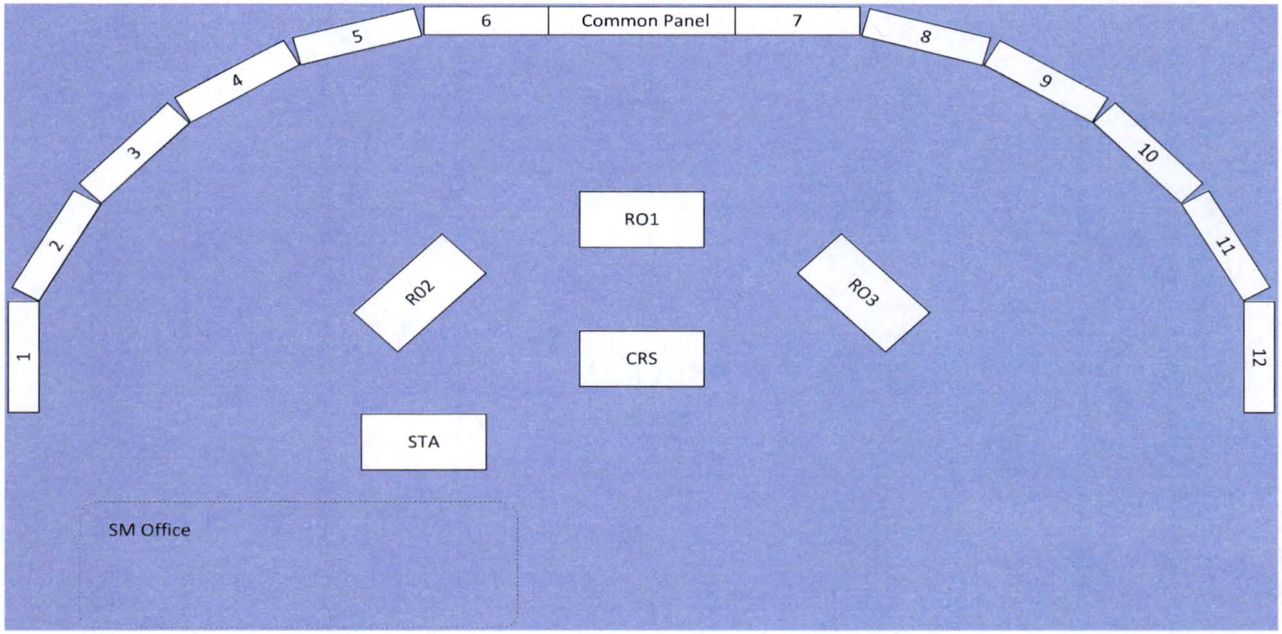
Plant Operations

Concept of Operations

- Simple and robust plant design with passive safety systems
  - Operators manage comparatively fewer and simpler systems
  - Design limits the severity, possibility, and progression rate of accidents
- State-of-the-art instrumentation & controls
  - Computer based measurements, analysis, and control of nuclear processes
  - Affords implementation of more automation
- Human factors engineering
  - Utilizing human factors engineering, operators and automatic agents can form an effective team for controlling the nuclear process
  - With the support of automatic agents, one human agent can be responsible for more than one power production unit

Control Room Simulator

Concept of Operations



Five crews of ten operators

- three senior reactor operators
- three reactor operators
- four non-licensed operators

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Control Room Simulator

Purpose of control room simulator

- Demonstrate
  - multiple units and shared system can be monitored and controlled from one control room
  - one reactor operator can monitor and control multiple units
  - concept of operations for a multiple-unit plant during normal operations, startup, shutdown, refueling, and upset conditions
- Inform the NuScale plant design
  - validate design features
  - validate system response
  - develop and test I&C logic

NuScale Power™

Main Control Room (MCR) Design

- Located in physically separate {{ }}2(a),(c) control building
- Single MCR for all 12 power modules
- Separate set of display screens for each module
- Common central display panel for all modules
- Normal MCR HVAC isolates on radiation signal or loss of AC power
- Backup diesel generators designed to power MCR HVAC on a loss of AC power
- Pressurized air in subterranean level of control building provides 72 hours of habitability along with below grade location of MCR (analogous to AP 1000 design)
- Underground tunnel with sealed doors from the MCR provides access to remote shutdown station in the reactor building

Reactor Building Crane (RBC)

RBC Function

- Lifts and moves the complete power module during refueling
- Lifts and moves upper CNV while assembling and disassembling the power module during refueling and initial receipt of new modules
- Two lift heights with the core inside the module
  - {{ }}2(a),(c) during movement in and between the module operating bays
  - {{ }}2(a),(c) above the UHS pool floor when placing the module into the lower CNV flange tool

NuScale Power™

RBC with Module Lifting Adapter

{{

}}2(a),(c)

NUSCALE POWER™

Single Failure Proof Main Hoist

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}}2(a),(c)

Main Hoist Reeving System

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}}2(a),(c)

NuScale Power™

Module Lifting Adapter (MLA)

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}}2(a),(c)

Power Module Lift Points

{{

}}2(a),(c)

NuScale Power™

MLA Paddles and Lift Points

{{

}}2(a),(c)

RBC Failure Probability

- Designed as single failure proof crane compliant with:
  - ASME NOG-1-2004
  - NUREG-0554
  - NUREG-0612
- PRA performed on reactor building crane included
  - initiating event identification
  - failure modes and effects analysis
  - structural failures
  - abnormal lifting events
  - human errors of commission

{{

}}2(a),(c)

Probabilistic Risk Assessment (PRA)

NuScale PRA

- All operating modes and all hazards
  - full-power internal-events
  - low-power/shutdown
  - internal fire
  - internal flood
  - high winds and hurricanes
  - other external events (seismic margins analysis)
- Single module CDF and LRF
  - Level-2 model very simple (CNV bypass/isolation)
  - simplified/focused look at multi-module hazards
    - only looking at dependencies

Perspective on Probabilities

NRC CDF Safety Goal Surrogate	1E-4 /reactor-yr
Typical large operating plant CDF	1E-4 to 1E-6 per reactor-yr
Typical ALWR CDF	~1E-7 per reactor-yr
NuScale total internal events full power CDF (May 2016 PRA*)	{{
Highest individual NuScale internal events CDF sequence (May 2016 PRA)	
NuScale large release frequency (LRF) (May 2016 PRA update)	
Highest individual NuScale internal events CDF sequence (May 2016 PRA)	}}2(a),(c)

*Since 2011, the NuScale internal events PRA has been updated 23 times

Initiating Event Contributors to PRA CDF

{{

}}2(a),(c)

Initiating Event Contributors to PRA LRF

{{

}}2(a),(c)

Spent Fuel Pool and Storage Rack Design

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Spent Fuel Pool Safety

Increased cooling capacity

- More water volume for cooling per fuel assembly than current designs
- Redundant, cross-connected reactor and refueling pool heat exchangers provide full back-up cooling to spent fuel pool

External coolant supply connections

- Auxiliary external water supply connections are easily accessible to plant personnel and away from potential high radiation zones

Below ground

- Below ground spent fuel pool is housed in the Seismic Category I reactor building
- Pool wall located underground is shielded from tsunami wave impact and damage

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Spent Fuel Storage Rack Design Parameters

- All primary stresses in the racks satisfy the limits in Section III, Subsection NF, Class 3 of ASME BP&V Code
- Fuel racks are able to store fuel with 5 weight percent maximum enrichment while maintaining neutron multiplication factor (k_{eff}) less than 0.95
- Fuel assembly drop event evaluation shows that the criticality geometry of the rack will not be compromised
- Demonstrate decay heat removal such that no nucleate boiling occurs

Spent Fuel Storage Structure

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Spent Fuel Storage Structure

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Spent Fuel Storage Structure

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NuScale Power

Spent Fuel Storage Design

- Racks are designed to Seismic Category I requirements and meet the stress limits of ASME BPVC Section III, Division I, Subsection NF-Supports, Class 3
- Design, fabrication, and examination of racks are in accordance with guidance from NF-3000 (design), NF-4000 (fabrication) and NF-5000 (examination)
- Spent fuel racks are designed to withstand normal and postulated dead loads, fuel drop loads, loads resulting from thermal effects, and loads caused by an SSE

Ultimate Heat Sink Reactor Building Pool

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Ultimate Heat Sink Pool

- Ultimate heat sink (UHS) is a highly engineered and protected feature in the NuScale design
  - Totals {{ }}2(a),(c) (includes spent fuel pool)
- UHS (reactor pool) is housed in seismic category 1, aircraft impact resistant reactor building
- {{ }}2(a),(c) weir wall separates reactor pool from spent fuel pool
  - {{ }}2(a),(c) gallons above the top of the weir
  - SFP has {{ }}2(a),(c) gallons from top of weir to top of fuel with {{ }}2(a),(c) gallons total in SFP below top of weir
- Reactor/spent-fuel-pool very robust
  - 0.5g SSE vs 0.3g maximum for operating fleet
  - {{ }}2(a),(c) thick concrete floor and {{ }}2(a),(c) thick concrete walls (all are lined with stainless steel)
- Subsurface design makes pool drain down an incredible event
- No credible mechanism for draining either pool

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Reactor Building Pool

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Internal Pool Leak Realistic Flood Level

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NUREG-2161 Leak Definitions

- Moderate leakage rate
  - A state with leakage from the bottom of the SFP, corresponding to through-wall concrete cracking at the bottom of the walls and tearing of the liner that propagates to an extent such that water leakage is controlled by the size of the cracks in the concrete
    - average leak rate ~1,500 gpm
- Small leakage rate
  - A state with leakage from the bottom of the SFP, corresponding to through-wall concrete cracking at the bottom of the walls and tearing of the liner that remains localized to the where the floor liner is attached to the SFP floor near the walls
    - average leak rate ~200 gpm

NuScale Sensitivity Study on Rx-Pool

- Preliminary analysis on time to air coolability of modules
  - Analysis performed in early 2014 (Rev B design) using MELCOR 1.8.6
- Sensitivity study to drain reactor pool (not SFP) over time

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- MELCOR used to simulate response of modules (all 12)

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Containment Pressure

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Comparison to a Large Passive PWR

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NuScale—AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Reactor Core, Primary Coolant and Relevant BOP Systems			
Core thermal power (MWth)	3,400	160	~5% of AP1000
Core decay heat (MWth) at 1 second 1 hour 1 day 1 month (30 days)	 212 47 23 8.5	{{	Less decay heat needs to be removed to protect the core from overheating and damage
Average core coolant flow rate (feet per second) at 100% power	16		Larger margin to flow induced vibration
Core pressure drop (psi)	40		Larger margin to cross flow induced vibration

}}2(a),(c),ECI

[1] Source: AP1000 DCD on NRC ADAMS web site
http://pbadupws.nrc.gov/docs/ML1117/ML11171A500.html

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NuScale–AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Reactor Core, Primary Coolant and Relevant BOP Systems			
Peak fuel centerline temperature (°F) at 100% power	4,700	{{	Larger fuel temperature margins to melting
Average core batch discharge burnup (MWD/MTU)	42,000		Lower fuel duty, smaller FP inventory
Peak fuel rod discharge burnup (MWD/MTU)	60,000		Lower fuel duty, smaller FP inventory
Total UO_2 in core (pounds)	211,588	}}2(a),(c),ECI	Smaller FP inventory; fewer fuel assembly moves during refueling
Number of fuel assemblies	157	37	
Active height of fuel (feet)	14	6.5	

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NuScale–AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Reactor Core, Primary Coolant and Relevant BOP Systems			
Number of hot legs and pipe ID (inches)	2 31	0 N/A	No external primary coolant piping
Number of cold legs and pipe ID (inches)	4 22	0 N/A	No external primary coolant piping
Pressurizer surge line pipe ID (inches)	18	N/A	Pressurizer internal to RPV; no surge line
Largest primary coolant system pipe size (inches)	31	{{ }}2(a),(c),ECI	No large or medium size pipe break LOCAs
Number of RCPs	4	0	No RCPs
Ratio of pressurizer steam space to core power (ft^3/MWth)	0.32	{{	Minimizes cycling components for pressure and level control, greater margin to trip setpoints
Ratio of pressurizer heater capacity to rated core thermal power (%)	0.047		Greater margin to trip setpoints
RCS liquid volume including pressurizer ($ft.^3$)	9,600		Relatively large RCS volume for small core
Ratio of RCS liquid volume to core power ($ft.^3$/MWth)	2.8	}}2(a),(c),ECI	Greater coolant thermal capacity margin
Uses Squib valves for post-accident cooling	Yes	No	No squib valves in NuScale design

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NuScale–AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Reactor Core, Primary Coolant and Relevant BOP Systems			
Insulation used on RCS components inside containment	Yes	No	Prevents insulation blockage of ECCS recirculation (GSI-191)
Number of residual heat removal pumps	2	None	Passive system; uses UHS pool and HX; no pumps
Number of auxiliary feedwater pumps	2	None	Uses existing MFW pumps; lower flow for MFW/AFW
Hydrogen used in main power generator	Yes	{{	}}2(a),(c),ECI
Number and rating of non-safety backup diesel generators	2 4 MW(e) each	{{ }}2(a),(c),ECI	Less need for backup electric power; more FLEX alternatives
Number of power operated relief valves on main steam line	1	{{	
Number of safety valves on main steam line	6		}}2(a),(c),ECI

NuScale–AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Fission product barriers			
Primary coolant system operating pressure (psia)	2,250	1,850	Reduced initial LOCA break flow
Containment operating pressure (psia) Containment design pressure (psia) Containment design temperature (^{o}F)	14.7 73.7 300	< 1.0 1,000 {{ }}$^{2(a),(c),ECI)}$	Minimizes GS-191 issues; improves LOCA steam condensation
Containment volume (ft.^{3})	2,060,000	{{ }}$^{2(a),(c),ECI}$	Sized so core does not uncover
Hydrogen igniters or recombiners used	Yes	{{	}}$^{2(a),(c),ECI}$
Containment spray system	Yes, but not safety-grade	No	Only natural processes are relied on for FP removal
Water depth in spent fuel pool (feet)	42.5	{{ }}$^{2(a),(c),ECI}$	More margin of water over the top of the SNF assembly

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NuScale–AP1000 Comparison

Parameter	AP1000[1]	NuScale	Impact on margin
Heat removal			
Total water inventory immediately available for core heat removal (ft.^{3}) Ft.^{3} coolant/MWth	82,000 from all CMTs, accumulators, and IRWST 2.4	{{ }}$^{2(a),(c),ECI}$	Reactor module pool area excluding spent fuel and refueling pools; very large margin for decay heat removal
Number and name of external water sources for post-accident core and containment heat removal	2 core makeup tanks (CMTs) 2 accumulators 1 In containment refueling water storage tank	None	UHS reactor building module pool provides all post-accident core and containment heat removal

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NuScale–AP1000 Comparison

Design Basis Accident Transient Response			
LOCA PCT (°F)	1,837 (Best Estimate)	{{ }}2(a),(c),ECI	No core uncovery for all DB LOCAs
Locked Rotor PCT (°F)	1,900	N/A	No RCPs
Peak calculated containment pressure (psia)	73	{{ }}2(a),(c),ECI	High pressure containment
Inadvertent ECCS operation can increase RCS pressure	Yes	No	ECCS operation depressurizes RCS
Beyond Design-Basis-Accident Transient Response and PRA			
Mean internal events core damage frequency (per reactor-year)	2.41×10^{-7}	{{ }}2(a),(c),ECI	Smaller CDF
Core and Containment Station Blackout Coping Time w/o electric power	72 hours	Indefinite	UHS requires no external water source or power indefinitely

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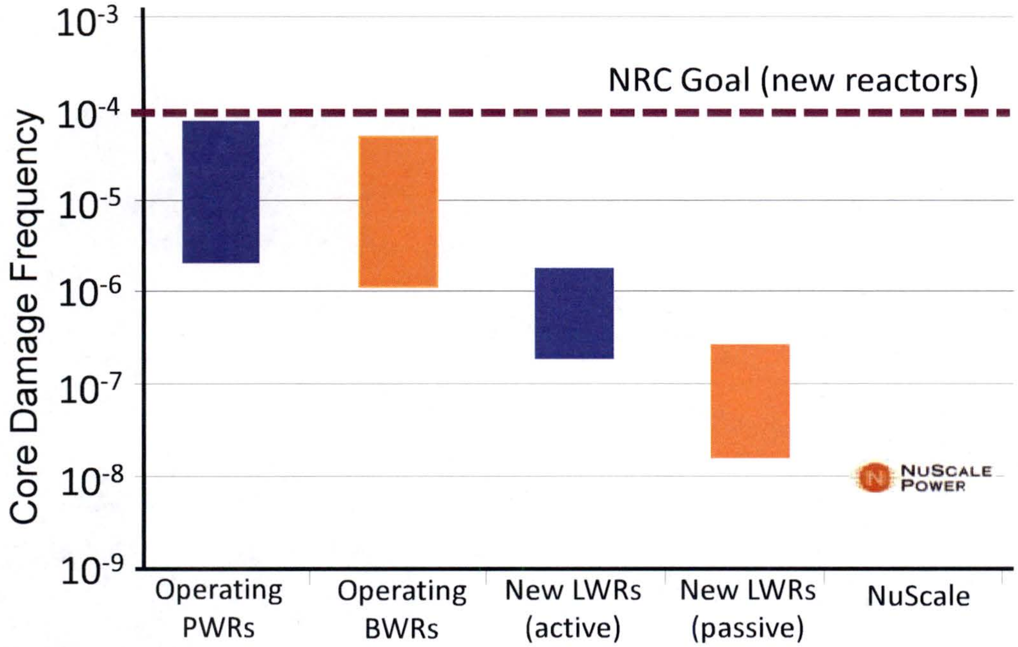
Accident Response

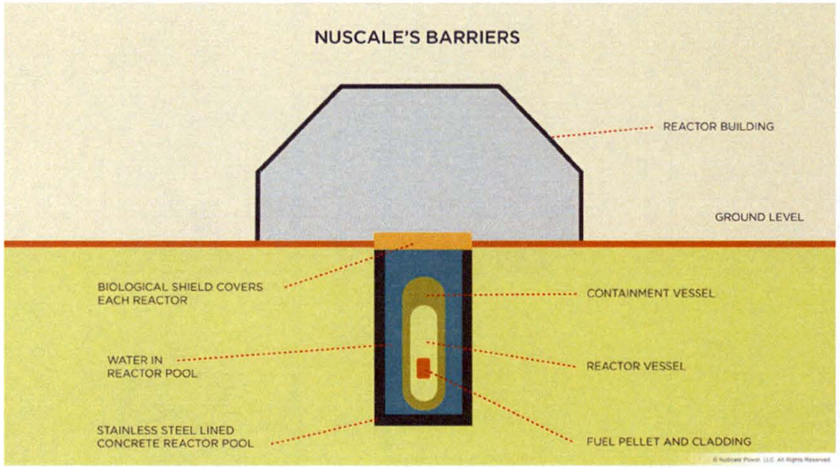
Response to Classic Accident Initiators

Design Basis Accident	NuScale Response
Steam system pipe break	**Reduced consequences** from lower energy release due to low steam generator inventory
Feedwater system pipe break	No change
Reactor coolant pump shaft failure	**Eliminated** with use of natural circulation of primary coolant
Control rod ejection accident	No change
Steam generator tube rupture	**Reduced likelihood** because tubes are in compression (shell-side primary flow)
Large break loss-of-coolant accident	**Eliminated** by use of integral design
Small break loss-of-coolant accident	**Reduced consequences** due to no heatup of fuel (already in natural circulation)
Design basis fuel handling accident	**Reduced consequences** due to smaller source term in half-height assemblies

Reducing Plant Risk

Risk = (frequency of failure) X (consequences)





*Probability of core damage due to NuScale reactor equipment failures is **1 in 100,000,000 years***

Small Break LOCA (SBLOCA)

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SBLOCA

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}}2(a),(c),ECI

NuScale Power™

SBLOCA

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SBLOCA

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SBLOCA

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Design Basis SBLOCA Results

SB LOCA Break Location	Credit DHRS	NRELAP Calculated PCT (^0F)
2-inch CVCS Letdown Line	Yes	<700
2-inch CVCS Charging Line	Yes	<700
2-inch Pressurizer Spray Line	Yes	<700
2-inch Pressurizer Vent Line	Yes	<700
Inadvertent RVV or RRV Opening	Yes	<700
2-inch CVCS Letdown Line	No	<700
2-inch CVCS Charging Line	No	<700
2-inch Pressurizer Spray Line	No	<700
2-inch Pressurizer Vent Line	No	<700
Inadvertent RVV or RRV Opening	No	<700
Normal Full Power Operation	N/A	<700

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ATWS Response

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ATWS Chronology of Events

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}}2(a),(c),ECI

Energy Balance (0 to 300 seconds)

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}}2(a),(c),ECI

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Energy Balance (0 to 3 hours)

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Core Reactivity

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}}2(a),(c),ECI

Peak Cladding Temperature

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Core Subcooling Margin

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RPV Pressure

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RCS Flow Rate

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ATWS Scenario Summary

- ATWS is a benign event for the NuScale design
- Natural circulation of primary system is self-limiting on core reactivity

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Early Core Damage LRF PRA Sequence

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RPV and CNV Pressure

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Collapsed Liquid Levels

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}}2(a),(c)

Power Module Coolant Distribution

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Sequence of Events

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CNV Short Term Pressure Response from RCS "Breaks"

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CNV Pressure Response after ECCS Initiates

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System Energy Balance After A CVCS Break in the CNV

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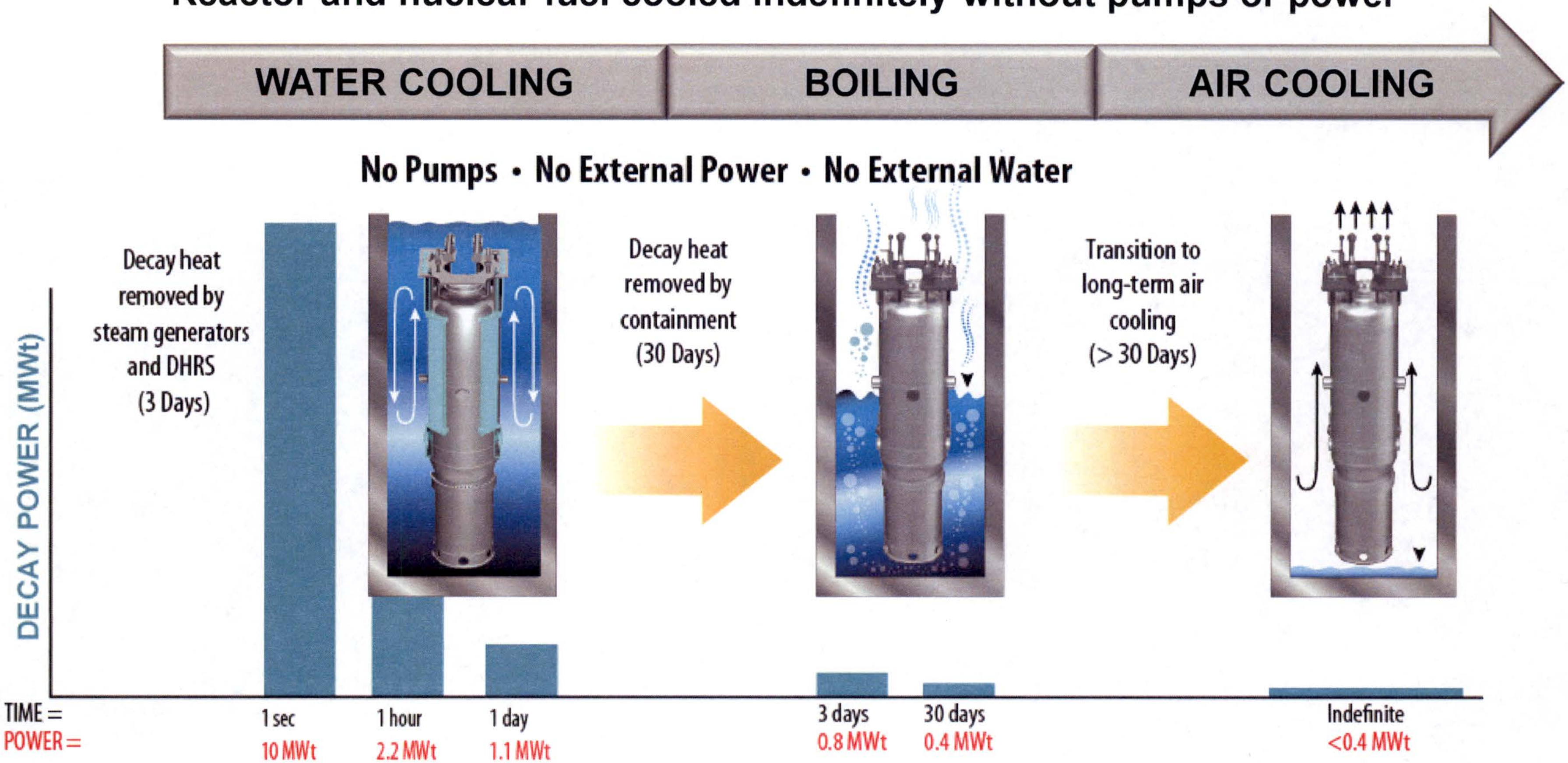
NuScale Power™

Extended Loss of All Power

Stable long-term cooling under all conditions
Reactor and nuclear fuel cooled indefinitely without pumps or power



** Based on conservative calculations assuming all 12 modules in simultaneous upset conditions and reduced pool water inventory (more realistic calculations show >>30 days for boil-down of pool)*

Extended Loss of Power Pool Level

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Revision: 0

Extended Loss of Power Radiation Doses

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DCA Test Program

Reactor Qualification Test Plan

- Summarizes test programs planned in support of design certification, FOAKE and product realization Projects
- Initial release July 2013; Latest release January 2016
- Includes individual test descriptions and identifies applicable requirements, including:
  - organization sponsoring test program
  - primary source requirements
  - applicable quality assurance requirements,
  - applicable references and various NuScale internal tracking codes
- Test were identified and prioritized using a risk-based process. Test required for DCA were given top priority.
- Updated periodically

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NuScale Reactor Qualification Test Plan
PL-1103-3463
(49 Test Programs)

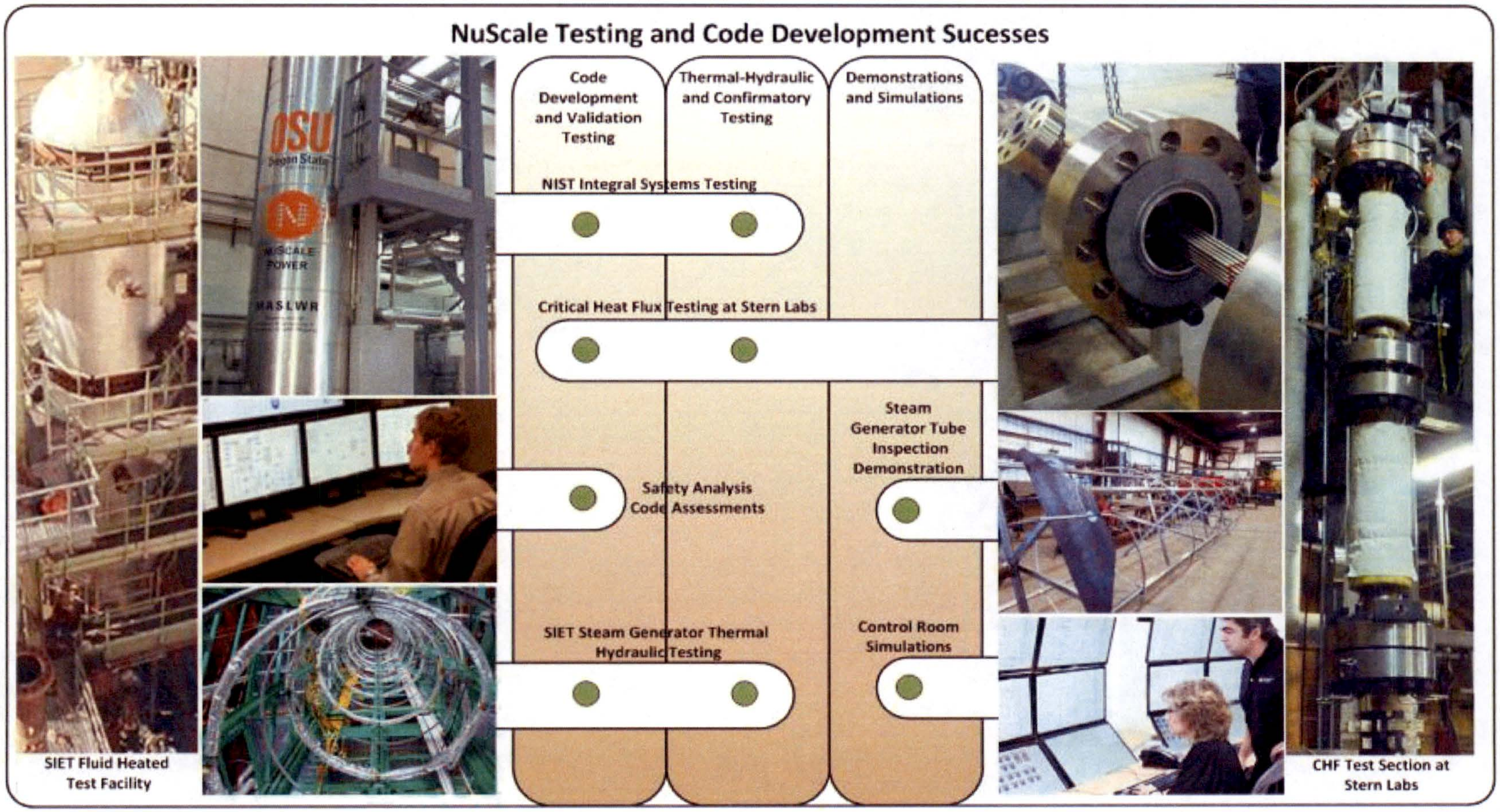
INCLUDING:

- ❑ Nuclear Fuel Mechanical
- ❑ Nuclear Fuel Thermal Tests
- ❑ Reactor Vessel Internals Handling
- ❑ Control Rod Drive Mechanism Tests
- ❑ Steam Generator Mechanical
- ❑ Steam Generator Thermal Tests
- ❑ RXM Manufacturing Demo
- ❑ Valve Testing
- ❑ Instrument Sensors
- ❑ Module Protection System Prototype
- ❑ Upper Module Mock-up
- ❑ Module Assembly Tools
- ❑ Integral System Tests
- ❑ Advanced Manufacturing
- ❑ Component Seismic Tests

This document was a collaborative effort with input from across NuScale and is central to our planning efforts and overall success

NuScale Reactor Qualification Test Plan

NuScale Reactor Qualification Test Plan outlines design certification and FOAKE projects for reactor safety code development, validation, reactor design, and technology maturation to reduce first-of-a-kind (FOAK) design risk.



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NuScale Reactor Qualification DCA Test Programs

- DCA tests consists of nine test programs with an approximate cost of $40M
- All test data needed to support the DCA will be completed prior to submittal in 2016. The NuScale test programs provide support for:
  - Chapter 3 – Design of Structures, Systems, Components and Equipment
  - Chapter 4 – Reactor
  - Chapter 5 – Reactor Coolant System and Connecting Systems
  - Chapter 15 – Transient and Accident Analysis
  - Chapter 19 – Probabilistic Risk Assessment
- NRC Audits/Inspections
  - Stern Laboratory, March 2013 (Inspection)
  - SG (SIET TF1), December 2013 (Inspection)
  - NIST-1, August 2014 (Inspection)
  - CHF (AREVA-Germany), (Audit) May 2016
  - Fuel Seismic (AREVA-Richland, WA), (Audit) May 2016
  - NIST-1, (Audit), June 2016

Testing Capabilities/Objectives

- System interaction testing
  - reactor vessel and containment coupling (pressures and levels)
  - DHRS characterization
- NRELAP5 code validation
  - LOCAs
  - long-term cooling
  - high-pressure condensation
  - cooling pool convection
  - non-LOCA transients (scram, overheating, overcooling, NC operation)
  - CVCS line break
- Safety methods development
  - evaluation model nodalization
  - evaluation model features
  - emergency procedures
- Simulator validation

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DCA Testing Progress

Test/Demonstration Program	Test Facility	Status
Critical Heat Flux Test – Initial Fuel Design	Stern Lab, Canada	Completed
Steam Generator Tube Inspection Feasibility Study	Corvallis, Oregon	Completed
SIET TF1; 3-Coil, Full-Length, Electrically Heated Steam Generator	SIET, Piacenza, Italy	Completed
SIET TF2; 252-Coils, Full-Length, Prototypic Fluid Heated Steam Generator	SIET, Piacenza, Italy	Completed
Upper Module Mock-up	OIW, Vancouver, WA	Completed
Fuel Mechanical and Hydraulic	Richland, WA	Completed
NIST-1 Facility; Integral System	OSU, Corvallis, OR	Critical Path and Stability Testing Complete
CRA and Drive Shaft Alignment and Drop	Erlangen, Germany	Under Construction
Steam Generator Flow Induced Vibration	Erlangen, Germany	Under Construction
CRA/CRAGT Flow Induced Vibration	Erlangen, Germany	Under Construction
Critical Heat Flux– NuScale Fuel Design w/AREVA HMP/HTP grids	Karlstein, Germany	In Test
Steam Generator Orifice Hydraulic Testing	Alden, Mass.	In Test
Percentage of Required DCA Testing Completed:	{{	
Total DCA Testing Expenditures:	}}2(d)	

NuScale Integral System Test Facility (NIST-1)

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NIST-1 Facility

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NIST-1 Containment and Cooling Pool

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NIST-1 Testing and Code V&V Support

- **System interaction testing** {{
  - Reactor vessel and containment coupling (pressures and levels)
  - DHRS characterization
- **Planned NIST-1 matrix tests for NRELAP5 validation:**
  - LOCAs
  - long-term cooling
  - high pressure condensation
  - cooling pool convection
  - non-LOCA transients
  - CVCS line break
- **Safety methods development**
  - evaluation model nodalization
  - evaluation model features
  - emergency procedures
- **Simulator validation**
- **Status:**
  - Testing in progress
  - NRC inspection week of August 24, 2015

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NIST-1 Test Facility

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Comparison of NRELAP-5 to NIST Data

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NRELAP5 Comparison of NPM and NIST-1 CVCS Line Breaks

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NRELAP5 Comparison of NPM and NIST-1 CVCS Line Breaks

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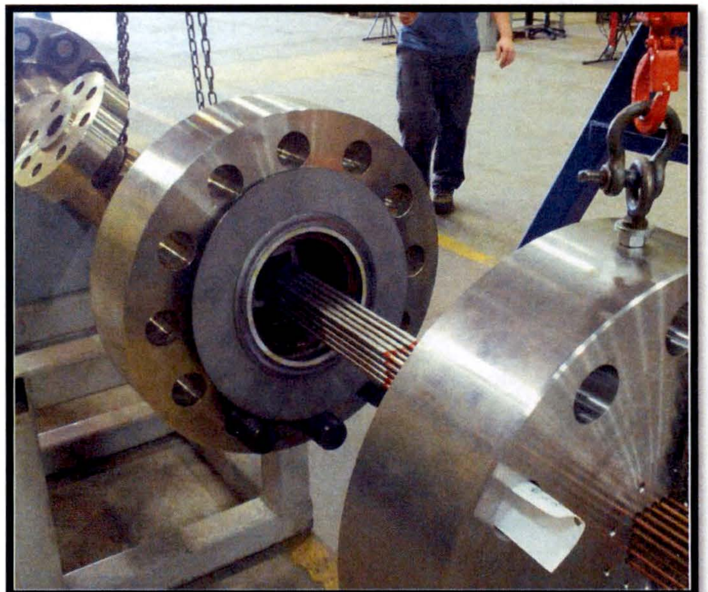
Fuel Critical Heat Flux Test at Stern Labs Canada

Objective: obtain prototypic full-scale thermal-hydraulic data for validating our sub-channel analysis code for NRC design certification.

Status:

- completed test program in March 2013
- testing provided CHF, pressure drop and thermal mixing data
- preliminary CHF correlation developed
- results indicate fuel design safe for NC flow
- CHF test report issued to the NRC



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Steam Generator Tube Inspection Demonstration

Objective: Design and demonstrate a proof-of-concept inspection system for the helical-coil steam generator using available supplier tooling

Status:

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- successfully completed proof-of-concept test program in Jan 2013
- only minor tooling modifications required
- testing showed that UT probe could be fully inserted / retracted from smallest and largest helical radius tubing
- results demonstrated feasibility of tube in-service inspection

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Separate-Effects SG Test at SIET Labs Italy

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Integral SG Test at SIET Labs Italy

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Seismic – FA Vibration and Damping

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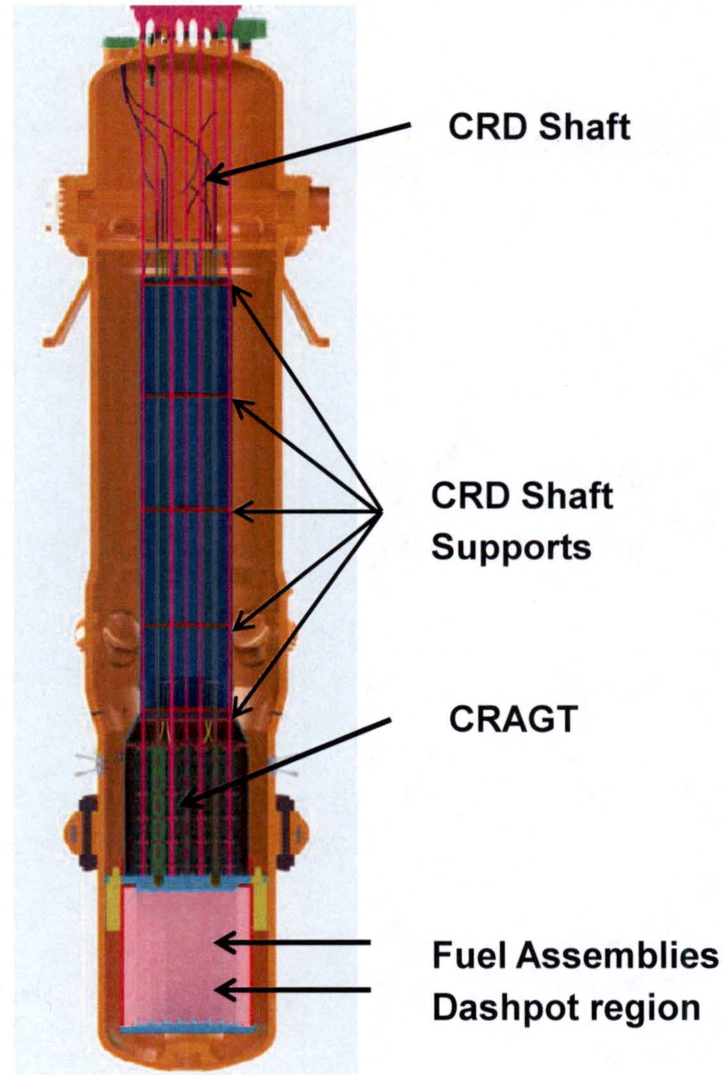
CRA and Drive Shaft Drop Alignment Test

Objectives:

- Demonstrate control rod assembly (CRA) insertion with drive shaft support, CRA guide tube (CRAGT) and guide card misalignment and fuel assembly distortion.
- Measure fuel insertion rate profile under different operational conditions.

Status of Phase 1 cold test:

- Test hardware delivered and being assembled
- Start of testing = April 2016



PM-0616-49510-NP
Revision: 0

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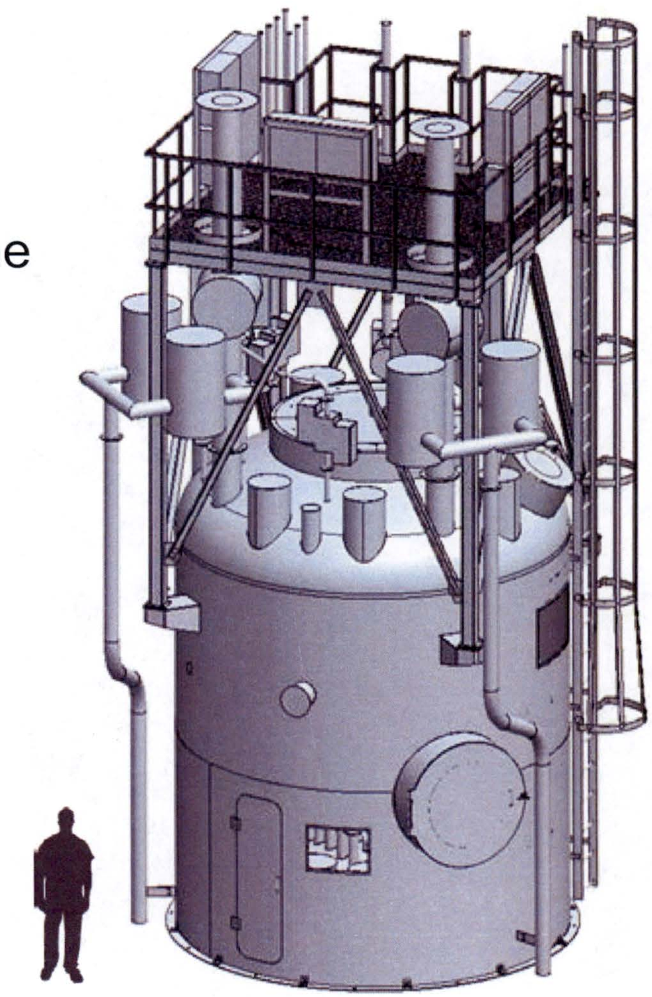
Upper Module Mockup

- A full scale mockup of the reactor module
- Built using non-prototypic material and material thicknesses
- Mockup geometry only includes the top of the RXM down to the elevation of the reactor vessel head
- All major components are mocked up including:
  - the upper portion of the containment vessel
  - major piping such as steam, feed water, and CVC
  - control rod drive mechanisms
  - major valves such as isolation valves and ECC valves
  - the RXM platform



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Upper Module Mockup

Supports final engineering, maintenance evaluations and inspection procedures

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Security by Design And Aircraft Impact Assessment

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Reactor Building Grade Level

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Security By Design

- Aircraft impact resistant reactor building
- Most safety-related components below grade
- Main control room below grade
- Spent fuel pool below grade
- Operating modules are submerged in the ultimate heat sink
- Passive safety systems
- No large bore piping containing reactor coolant
- No reliance on operator actions, electrical power, or additional water to maintain the reactor core or spent fuel safe

Proposed Security Staffing

- Potentially reduced security staffing numbers
  - Fewer targets
  - No required operator actions in design basis space
  - Reduced site footprint compared to larger reactors in operation
  - Reduce protected area access facility staffing due to reduced facility size
  - Reduced security response force due to incorporation of new technology into the protective strategy
  - Reduce security staffing through shared roles
    - Response team leader role covered by security shift supervisor, if always indoors
    - Protected and vital area/insider mitigation program tour officers role covered using camera monitoring
    - Vehicle escort officer could be any qualified plant person unless escorting hazardous material

Aircraft Impact Assessment (AIA)

- 10 CFR 50.150 AIA has been performed
- Non-SGI part of AIA will be in DCD Section 19.5
- Both the reactor building and power modules were considered in the assessment using NEI 07-13 methodology
- Assessment in two documents
  - structural
  - fire and shock
- No deviations from NEI 07-13, but different component names were used (e.g., reactor building crane instead of polar crane)

Fire Protection

Fire Protection Design and Features

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Radiological Protection

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Reactor Module Bioshield Design

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Reactor Building Dose Rates

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Spent Fuel Pool Dose Rates

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The NuScale DCA

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NuScale DCA Content

Part 1 General and Financial Information

Part 2 DCD Safety Analysis Report

- Tier 1
- Tier 2

Part 3 Environmental Report

Part 4 Technical Specifications

Part 5 Emergency Plan*

Part 6 Security Plan*

Part 7 Exemptions, Departures, and Variances

Part 8 License Conditions; ITAAC

Part 9 Withheld Information (SUNSI)

Part 10 Quality Assurance Program Description

Part 11 Supplemental Information

* Not applicable for design certification

NuScale DCA Part 2 Tier 1 Content

Chapter 1 – Introduction

Chapter 2 – Unit Specific SSC Design Descriptions and ITAAC

Chapter 3 – Shared SSC and Non-SSC Design Descriptions and ITAAC

Chapter 4 – Interface Requirements

Chapter 5 – Site Parameters

NuScale DCA Part 2 Tier 2 Content

Chapter 1 - Introduction and General Description of the Plant

Chapter 2 – Site Characteristics and Site Parameters

Chapter 3 – Design of SSCs and Equipment

Chapter 4 – Reactor

Chapter 5 – Reactor Coolant System and Connecting Systems

Chapter 6 – Engineered Safety Features

Chapter 7 – Instrumentation and Controls

Chapter 8 – Electric Power

Chapter 9 – Auxiliary Systems

Chapter 10 – Steam and Power Conversion System

Chapter 11 – Radioactive Waste Management

Chapter 12 – Radiation Protection

Chapter 13 – Conduct of Operations

Chapter 14 – Initial Test Program and ITAAC

Chapter 15 – Transient and Accident Analyses

Chapter 16 – Technical Specifications

Chapter 17 – Quality Assurance and Reliability Assurance

Chapter 18 – Human Factors Engineering

Chapter 19 – Probabilistic Risk Assessment

Chapter 20 – Mitigation of Beyond-Design-Basis Events (Unique to NuScale DCA)

Chapter 21 – Multi-Module Design Features and Safety Basis Summary (Unique to NuScale DCA)

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Topical Report Submittals

	Title	Submittal Date	NRC Review/Development Status
1	Risk Significance Determination	July 2015	Accepted for review, NRC presented at the ACRS full committee meeting on May 5
2	Onsite DC Electrical System Safety Classification, Design, and Licensing Basis	October 2015	Accepted for review, awaiting first round of RAIs
3	EPZ Sizing Methodology and Application	December 2015	Accepted for review, awaiting first round of RAIs
4	Quality Assurance Program Description for the NuScale Power Reactor	March 2016	Final SER issued June 2, 2016
5	Accident Source Term Methodology	March 2016	Accepted review in progress
6	Highly Integrated Protection System Platform	March 2016	Accepted for review, RAIs expected in June 2016
7	AREVA Topical Report Applicability to NuScale Design	June 2016	Report development on track for timely submittal
8	Nuclear Analysis Codes and Methodologies	June 2016	Report development on track for timely submittal
9	Steady State Core Thermal-Hydraulics and Primary System Stability	July 2016	Report development on track for timely submittal
10	LOCA Evaluation Model	July 2016	Report development on track for timely submittal
11	Critical Heat Flux Correlation	September 2016	Report development on track for timely submittal
12	Subchannel Methodology	September 2016	Report development on track for timely submittal
13	Non-LOCA Methodologies	September 2016	Report development on track for timely submittal

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Planned DCA Exemption Requests

Exemption	Affected DCA Tier 2 SAR DCD Chapters
Combustible Gas Control 10 CFR 50.44(c)	6.2.5 and 6.3
Containment Heat Removal Testing 10 CFR 50, App. A, GDC 40	6.2.2.1
Containment Integrated Leak Rate Testing 10 CFR 50, App. A, GDC 52 and App. J	6.2.6
Containment Isolation 10 CFR 50, App. A, GDC 55, GDC 56, GDC 57	6.2.4
Offsite Power 10 CFR 50, App. A, GDC 17	Chapter 8
Control Room Staffing 10 CFR 50.54(m)(2)(i) and (iii)	13.1 and 18.5
Reactor Coolant System Venting 10 CFR 50.46a and 50.34(f)(2)(vi)	5.4.3
Reactor Coolant System Makeup 10 CFR 50, App. A, GDC 33	9.3.4
ECCS Evaluation Models 10 CFR 50, App. K	6.3
ATWS (Auxiliary Feedwater/ Turbine Trip) 10 CFR 50.62 (c)(1)	7.1 and 15.8.2

DCA Design and Analysis Computer Codes

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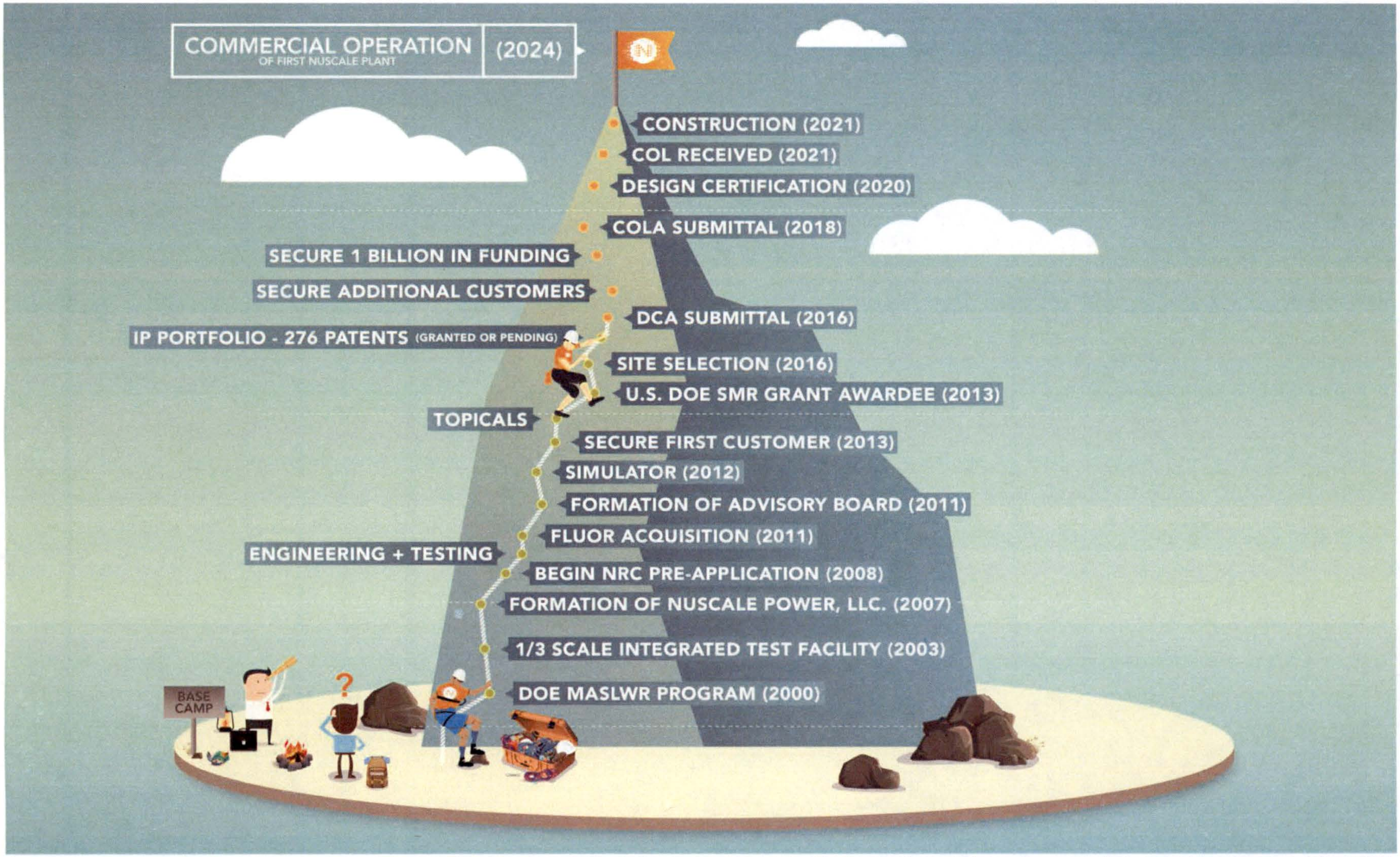
}}2(a),(c)

[1]NuScale version of existing computer code

Blazing the Trail to Commercialization



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